A SUMMARY OF MEASURED DELAYED NEUTRON GROUP PARAMETERS

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Abstract -The experimentally-measured delayed neutron parameters for 20 different fissionable isotopes are summarized. The decay curves measured for each isotope are compared as a function of the incident neutron energy inducing the fissions. Based on these comparisons, it is concluded that the results are quite wide spread and that further experimental work should be performed to clearly identify the most accurate sets of delayed neutron parameters for use in reactor applications.

INTRODUCTION

The reactivity of a reactor system is an inferred quantity. That is to say, it is not a primary quantity that can be directly measured. Instead, reactivity is always inferred from the measurement of some other quantity, such as an asymptotic period, or the alpha obtained during a Rossi-alpha measurement. In the case of a period measurement, the reactivity is inferred from the inhour equation in which the experimenter assumes a set of delayed neutron parameters and a neutron generation time. When determined in this fashion, the accuracy of the inferred reactivity is highly dependent on the accuracy and/or applicability of the assumed set of delayed neutron parameters used in the inhour equation.

There have been several in-pile techniques that have been developed over the years to test the accuracy of a given set of delayed neutron parameters. One of the earliest techniques was developed by Hansen (1951). In this technique, a series of small, equal reactivity additions are sequentially introduced into a reactor system. At each new reactivity configuration, the corresponding asymptotic period is measured. The reactivity inferred from the inhour equation is then calculated using an assumed set of delayed neutron parameters. If the delayed neutron parameters are applicable to that system, then the reactivity should theoretically increase in a linear fashion with each additional reactivity addition. For example, if each reactivity perturbation is \$0.05, then the reactor should be \$0.05 supercritical after the first perturbation, \$0.10 after the second perturbation, \$0.15 after third perturbation, etc. Hansen performed this technique on the Topsy reactor, which is a tuballoy-tamped oralloy sphere, in order to test the accuracy/applicability of the delayed neutron parameters measured by Hughes et al. (1948), and de Hoffman et al (1948). He found that Hughes' delayed neutron set predicted a linear behavior up to about \$0.50, whereas de Hoffman's delayed neutron set was only linear through approximately \$0.15. Hence, Hansen concluded that Hughes' delayed neutron parameters were more applicable for Topsy than those of de Hoffman, but were not perfect. Hughes's set

began to show significant non-linearity above \$0.50, indicating that the short-lived delayed neutron groups were not adequately characterized.

There is also new experimental evidence that suggests that reactivities inferred from the inhour equation using the currently accepted 6-group delayed neutron parameters of Keepin et al. (1957) might be biased by 3 to 4% in uranium systems. One example of this experimental evidence was obtained from an experiment performed on the University of Arizona's TRIGA reactor (Spriggs and Doane, 1993). In this experiment, the initial inverse periods, ω, corresponding to a series of superprompt-critical bursts were measured using low-noise fission chambers. It is well known that in small, tightly-coupled systems, ω varies as a linear function of reactivity, $\rho_{\$}$. That is,

$$\omega = \frac{\beta}{\Lambda}(\rho_{\$} - 1) \qquad \text{for } \rho_{\$} > \$1. \tag{1}$$

where $\rho_{\$}$ is the system reactivity (in dollars), ω is the asymptotic inverse period, Λ is the neutron generation time, and β is the effective delayed neutron fraction. Equation (1), of course, intersects the reactivity axis at \$1. However, as noted in the aforementioned experiment, when the measured inverse periods were plotted as a function of the *indicated* reactivity worth of the burst rod (which was calibrated using positive periods), the curve did not intersect the reactivity axis at \$1. It was postulated that this deviation occurred because the burst rod was calibrated using an inappropriate set of delayed neutron parameters.

To demonstrate this effect, the burst rod was re-calibrated using three different sets of delayed neutron parameters. The resulting burst rod calibrations for these three different sets of parameters are shown in Fig. 1. As noted from Fig. 2, when ω was plotted as a function of the indicated reactivity worth using each of three burst rod calibrations, the intercept changed significantly. Using Keepin's delayed neutron parameters, the intercept was found to occur at \$1.04; using the theoretical delayed neutron parameters found in ENDF/B-VI, the intercept occurred at \$0.93; and, using the in-pile measured delayed neutron parameters for this system (Spriggs, 1993), the intercept occurred at \$1.01. These results suggest that Keepin's 6-group delayed neutron parameters overestimated reactivity for this system by approximately 4%, ENDF/B-VI's theoretical delayed neutron parameters underestimated reactivity for this particular system by approximately 7%, and the measured in-pile delayed neutron parameters for this system overestimated the reactivity by approximately 1%.

In another experiment performed on the University of Arizona's TRIGA reactor, a series of negative period measurements were performed. The inferred reactivities corresponding to these measurements were then computed using the same three sets of delayed neutron parameters mentioned above. These results indicated that there could be large differences for the inferred reactivity depending on which set of delayed neutron parameters was assumed in the inhour equation. For example, the reactivity inferred from a measured period of -85 s corresponded to -\$0.789 when Keepin's parameters were assumed, and -\$0.407 when the delayed neutron parameters in ENDF/B-VI were assumed. In comparison, the measured system reactivity corresponding to a -85 second period was found to be approximately -\$0.764 based on an in-pile measurement of the delayed neutron parameters. These results suggest that Keepin's parameters overestimated negative reactivities by 3.3%, whereas, the ENDF/B-VI parameters underestimated negative reactivities by as much as 47%.

Although the discrepancies noted in the two experiments mentioned above seem to challenge the validity of the ENDF/B-VI recommended delayed neutron parameters and, to a lesser extent, Keepin's 6-group parameters, the cumulative experimental evidence found in the literature is not yet sufficient to justify abandoning either model. Rather, these two experiments merely suggest that we need to re-examine our cur-

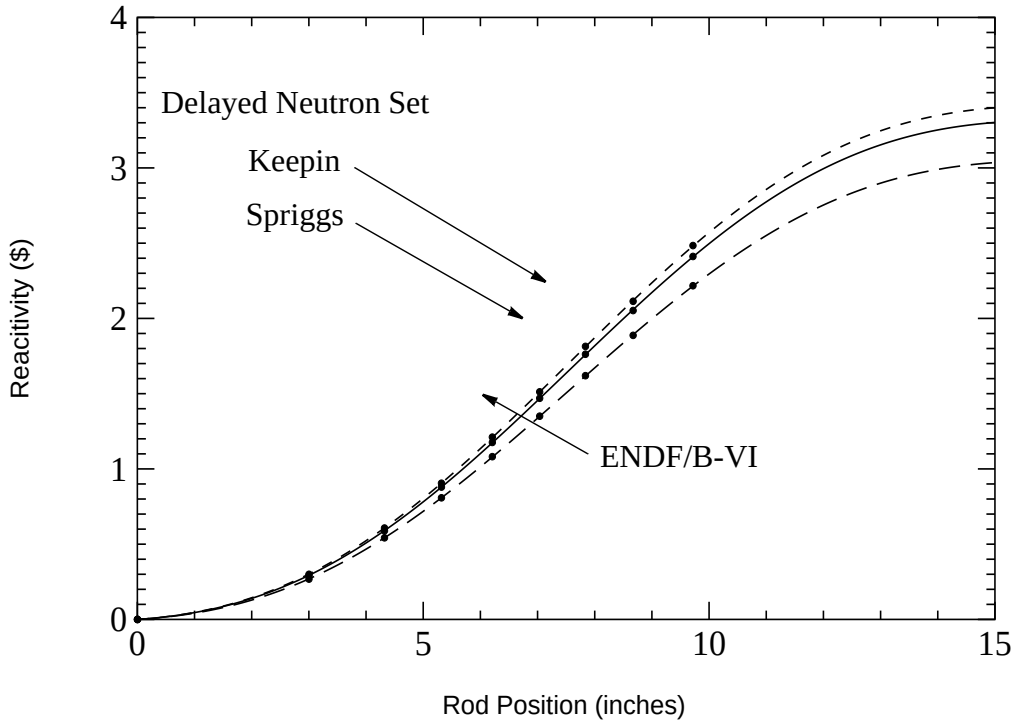

Fig. 1. University of Arizona's burst rod calibration curve assuming three different sets of delayed neutron parameters. The burst rod was calibrated in an incremental fashion using the Shim and Regulating rods to return the system to delayed critical after the reactivity worth of each section of the burst rod was inferred from a positive period measurement. (Data was not taken for rod positions greater than 10 inches since the burst rod is mechanically blocked at 10 inches to prevent inadvertent bursts greater than $2.50—the maximum allowable burst size for this system.)

rent delayed neutron data base and perform new experiments specifically designed to test the accuracy of that data base. Hopefully, this re-examination will result in an improved delayed neutron model that can predict a more accurate reactivity scale.[a]

In April 1997, an international workshop on delayed neutrons was held at the Institute of Physics and Power Engineering (IPPE) in Obninsk, Russia. The workshop was sponsored by the Nuclear Energy Agency's (NEA) working party on delayed neutrons (WPEC/SG6). The primary intent of this workshop was to review the current status of delayed neutron data and to propose new programs to improve these data for applications in reactor physics. As part of this international effort, these two authors were asked to perform a literature survey of measured delayed neutron group parameters to ascertain the extent of our data base.

[a.] Throughout the remainder of this work, we shall refer to the relationship between period (or inverse period) and reactivity as the *reactivity scale*. This relationship is highly dependent on the value of the delayed neutron parameters assumed in the inhour equation and is very important when measuring reactivity in operating systems.

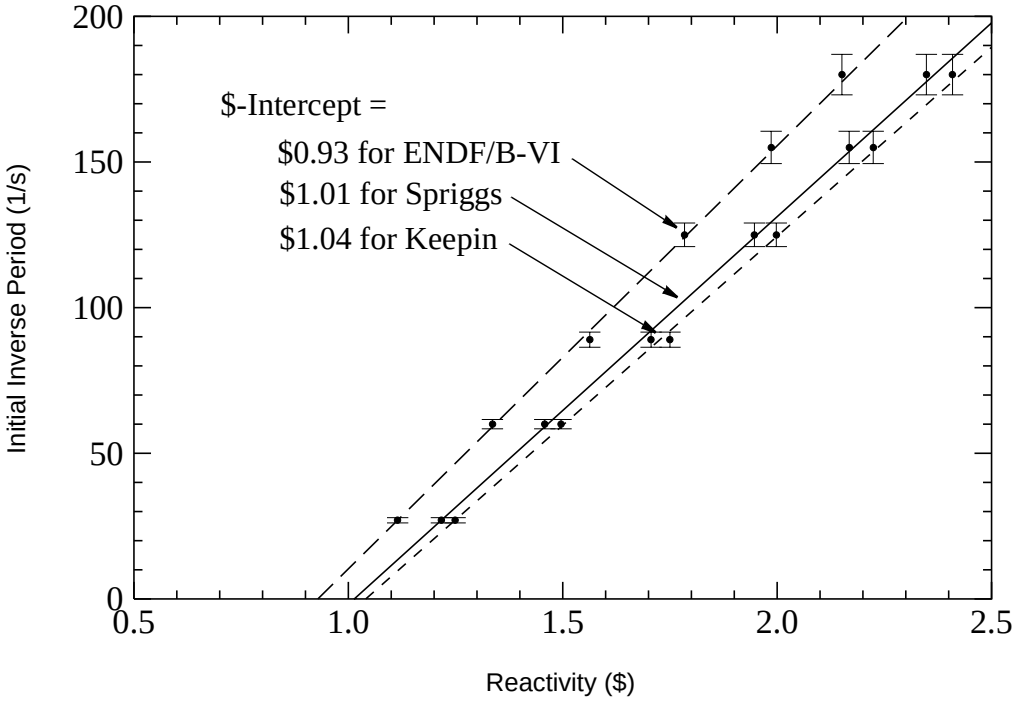

Fig. 2. Initial inverse period for a series of superprompt critical excursions vs. the indicated reactivity of a burst rod that has been calibrated by the period method assuming three different sets of delayed neutron parameters.

LITERATURE SURVEY

During our literature survey, we identified numerous articles on delayed neutron experiments. These experiments are listed in Table I. Several of these articles are of historical interest (1 through 5), and several review articles on delayed neutrons have been published during the past 50 years (6 through 16). To date, we have also identified 245 individual sets of delayed-neutron group parameters for 20 different fissionable isotopes (17 through 101). The articles from which these parameters were obtained are listed in chronological order in the Reference section of this report in order to maintain a better historical perspective of the work done in this area.

For purposes of dividing the various experimental results into categories characterized by the energy of the incident neutrons inducing the fission, we have defined four energy regimes—thermal, fast, transitional, and high. The *thermal* regime is $E < 10^{-6}$ MeV; the *fast* regime is $10^{-6} < E < 5$ MeV; the *transitional* regime $5 < E < 13$ MeV; and the *high* regime is $E > 13$ MeV. These regimes are based on the energies at which the second-chance fission and third-chance fission occur (see Fig. 3 and Fig. 4).

Table I. Bibliography of Delayed Neutron Experiments

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Table I. Bibliography of Delayed Neutron Experiments

No.	Reference
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94	V. M. PIKSAIKIN, private communication (October 2 and 10, 1997).
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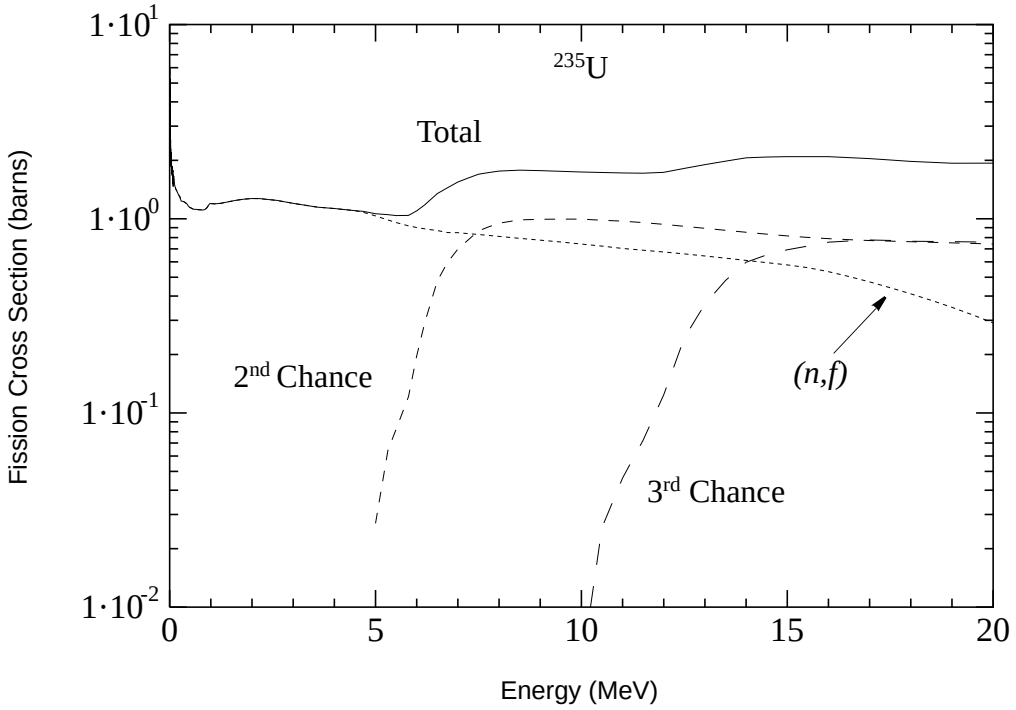

Fig. 3. Fission Cross Section for ^{235}U.

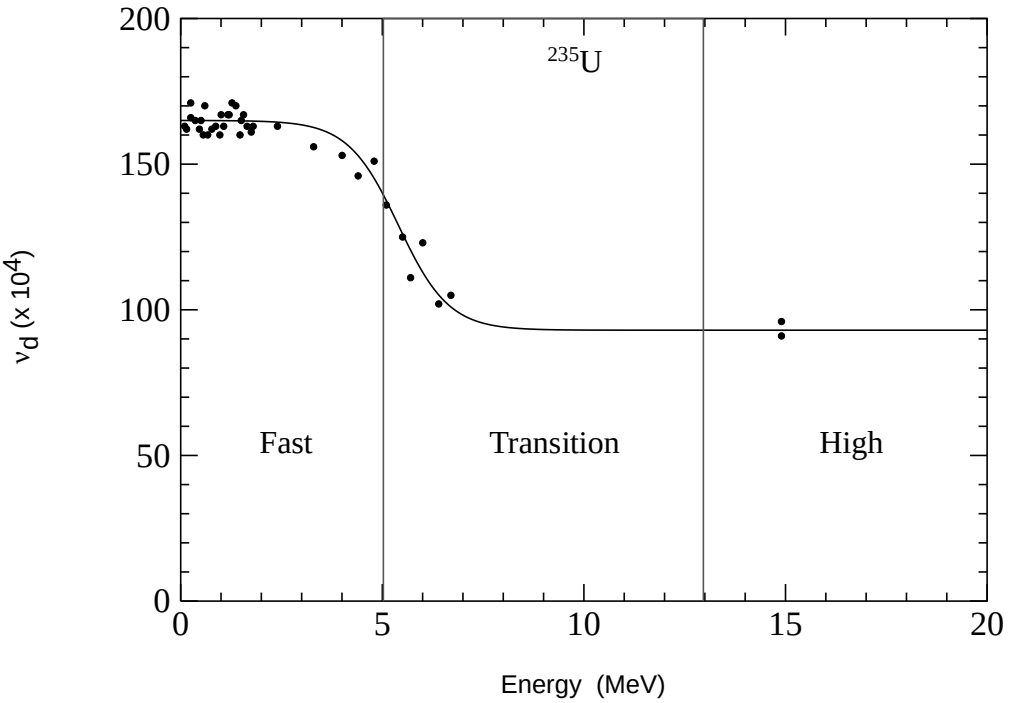

Fig. 4. Delayed Neutron Yield as a function of incident neutron energy.

When reviewing the literature, we tried to categorize each delayed neutron set in accordance to the energy of the incident neutron quoted by the experimenters (i.e., thermal, fast, high, etc.). However, in many cases, the experimenters did not state the energy of the incident neutron. When this occurred, we had to guess the incident neutron energy based on their description of the experiment, if any was provided. In most cases, it was felt that the incident neutron spectrum must have been, in all likelihood, a mixture of both thermal and fast fluxes and, as such, should not be categorized as either a thermal or a fast delayed neutron set. Nevertheless, sticking with the traditional thermal/fast categories for reactor physics application, we guessed where most of the fissions must have occurred using the energy-dependent fission cross section for each isotope as a guide. Accordingly, we categorized many of the delayed neutron sets as either thermal or fast despite the fact that they are probably best characterized as a mixed spectrum. We apologize in advance for any error in judgement.

RESULTS

Table II shows the distribution of measured delayed neutron group parameters for the 20 different isotopes reported in the literature as a function of the energy catagories defined above. Table III presents a brief summary of some of the pertinent information concerning the experimental conditions and the analysis techniques used to obtain the experimental results. In some cases, this information was not included in the original article. When omissions of this type occurred, we denoted this by placing a *question mark* in the appropriate column. In addition, we found references to several articles that have never been translated into English and, as such, could not be reviewed by these authors. A complete listing of the delayed neutron group parameters for these 245 sets can be found in Los Alamos National Laboratory report LA-UR-98-918 (web site URL: http://lib-www.lanl.gov/la-pubs/00393607.pdf).

COMPARISON OF DECAY CURVES

For comparison sake, we have plotted the delayed neutron decay curves for each of the isotopes that have been measured (see Fig. 5 through Fig. 42). As can be noted from these figures, there is considerable spread in the results. We speculate that there are several factors that have contributed to this spread.

First, the experimental techniques used to performed the decay curve measurements have improved dramatically since the first measurements performed in 1945. The sample sizes of the fissionable isotopes are larger, and are of higher purity. In addition, the neutron sources used to induce the fissions are stronger, and with each new generation of detectors and detector counting systems, the detector sensitivity has greatly increased while the dependence on incident neutron energy has greatly decreased. This has allowed for the measurement of more detailed decay curves showing less statistical variations. And finally, with the advent of multi-channel scalars, better time resolution has been achieved during the measurements.

Second, the analysis techniques used to analyze the decay curves have become more quantatitive with the advent of least-squares-fitting codes. Prior to the use of these codes, most of the decay curves were analyzed by graphical stripping techniques which can be somewhat subjective.

And third, and perhaps most importantly, as more and more experiments were performed, it became readily apparent that short sample transfer times were absolutely essential in resolving the short-lived delayed neutron groups. In most modern measurements, samples transfer times of less than a second or so are usually obtained. However, during some of the earlier measurements, sample transfer times as high as 30 seconds were reported. This difference in sample transfer time can have a very dramatic impact on the final results. When ever the transfer time is relatively long (i.e., > 1.0 seconds), the short-lived delayed neutron

groups (i.e., Groups 5 or 6 in the six-group model) cannot be resolved. Hence, the decay curves for those particular measurements show less of an initial drop and then decrease at a slower rate. Invariably, these curves are on the high side of the average curve. In contrast, the experiments in which the sample transfer times were very short (~0.1 seconds or less) had the best chance of observing the entire delayed neutron decay curve.

Table II. Summary of Out-of-Pile Delayed Neutron Experiments

Isotope	Thermal Spectrum	Fast Spectrum[a]	Transitional Energies[b]	High Energies[c]
Th-229	1			
Th-232		12	8	9
Pa-231		2		1
U-232	1			
U-233	6	4	6	11
U-235	16	22	7	8
U-236		1		
U-238		16	11	23
Np-237		29		
Pu-238	1	1		
Pu-239	6	6	5	10
Pu-240		3		
Pu-241	2	2		
Pu-242		1		2
Am-241	2	2		
Am-242m	1			
Am-243		3		
Cm-245	1			
Cf-249	1			
Cf-252	2[d]			

a. Thermal < E < 5 MeV
b. 5 MeV < E < 13 MeV
c. E > 13 MeV
d. Spontoneous fission.

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{229}Th	Gudkov et al. (1989)	86	50.2 μg Th	IRT Reactor, Russia	Thermal Spectrum	~1 – ? s	5	LSF
^{232}Th	Brolley et al. (1943)	18	11.3 Kg Th-Nitrate U < 0.1%	Univ. Chicago cyclotron: Be-target	Fast Spectrum	~0.5 – ? s	4	Graphical
^{232}Th	Cahn et al. (1945)	25	?	Argonne CP-3 reactor	Fast Component of Reactor Spectrum	0.4 – ?s	5	Graphical
^{232}Th	Creveling et al. (1949)	34	10.53 gms ^{232}Th metal	Ohio State Univ. cyclotron—^{7}Li(d,n)	Fast (max=24 MeV)	?	5	Graphical
^{232}Th	Sun et al. (1950)	35	4.5 Kg ^{232}Th oxide	Univ. Pittsburgh cyclotron—C, LiF, B_4C targets	14-29 MeV	? – 500 s	5	Graphical
^{232}Th	Rose & Smith (1957)	38	43.9 gms Th 100% ^{232}Th	ZEPHYR Reactor, Harwell	Fast Spectrum	~1 – 300 s	5	Graphical
^{232}Th	Keepin et al. (1957)	39	~2 – 5 gms Th 100% ^{232}Th	Godiva reactor, LASL	Fast Spectrum	0.05 – 330 s	6	LSF
^{232}Th	Maksyutenko (1958)	41	~180 gms Th	An accelerator—heavy ice target	2.4, 3.3, 15 MeV	~0.25–360 s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{232}Th	Hermann et al. (1965)	47	A few gms ^{232}Th metal	An accelerator	14 MeV	?	4	?
^{232}Th	Maksyutenko (1965)	50	?	?	1.6, 1.9, 2.2, 2.6 MeV	?	5	LSF
^{232}Th	Maksyutenko (1967)	52, 54	?	Van de Graaff generator Zr-deuterium target	5, 6, 6.2, 6.4, 6.6, 6.8, 7.25, 7.5, 7.75 MeV	?	5	LSF
^{232}Th	Hermann (1967)	53	Several gms Th metal	An accelerator– Be-D reaction	14 MeV	0.3 – 400 s	6	LSF
^{232}Th	Maksyutenko (1967)	54	?	?	5.0, 6.0, 6.2, 6.4, 6.6, 6.8 7.25, 7.7, 7.75 MeV	?	5	LSF
^{232}Th	Cox & Whiting (1968)	59	~45 gms Th	Van de Graaff, Li(p,n)	1.45, 1.50, 1.64 MeV	0.2 – 400 s	5	LSF
^{232}Th	Notea (1969)	62	?	An accelerator	14 MeV	?	5	?
^{232}Th	Brown et al. (1971)	71	~8 gms Th metal powder	S.A.M.E.S. neutron generator, ^{3}H(d,n)^{4}He	14.8 MeV	?	4	?
^{232}Th	Benedict et al. (1972)	75	?	?	14.8 MeV	?	6	LSF
^{232}Th	Waldo et al. (1981)	82, 83	A few μg/mg of chemically purified Th >99.5% ^{232}Th	Livermore Pool-Type, Thermal Reactor, LLNL	Fast Component of Reactor Spectrum	0.3 – ? s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{231}Pa	Chrysochoides et al. (1970)	70	5 mg Pa oxide in Cd container	Democritos Reactor, Greece	Above Cd Cutoff	30 – 320 s	3	LSF
^{231}Pa	Brown et al. (1971)	71	7.25 gms Pa_2O_5	S.A.M.E.S. neutron generator, $^{3}H(d,n)^{4}He$	14.8 MeV	?	4	?
^{231}Pa	Anousis et al. (1973)	77	A few mg Pa_2O_5 in Cd container	Democritos Reactor, Greece	Above Cd Cutoff	5 – 272 s	6	LSF
^{232}U	Waldo et al. (1981)	82, 83	A few μg/mg of isotopically purified U 99.99% ^{232}U	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Reactor Spectrum	0.3 – ?s	5	LSF
^{233}U	Cahn et al. (1945)	25	?	Argonne CP-3 Reactor	Thermal Spectrum	0.25 – 0.4 s	5	Graphical
^{233}U	Girshfeld (1955)	37	58.9 mg ^{233}U foil	? reactor	Thermal Spectrum	0.4 – 180 s	4	Graphical
^{233}U	Rose & Smith (1957)	38	10.0 gms ^{233}U	ZEPHYR Reactor, Harwell	Fast Spectrum	~1 – 300 s	5	Graphical
^{233}U	Keepin et al. (1957)	39	~2–5 gms U 100% ^{233}U	Godiva Reactor, Los Alamos	Thermal & Fast Spectra	0.05 – 330 s	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{233}U	Maksyutenko (1963)	45	3.63 gms ^{233}U	? accelerator	15 MeV	?	5	LSF
^{233}U	Maksyutenko (1967)	54, 56	?	Van de Graaff, Zr-Deuterium target	5.6, 6.0, 6.2, 6.4, 6.8, 7.25 MeV	?	5	LSF
^{233}U	Rambo (1969)	63	15.251 gms UO_2 98.33% ^{233}U	VPI Research Reactor, Virgina	Thermal Spectrum	~0.04–319 s	5	LSF
^{233}U	Onega et al. (1969)	64	15 gms ^{233}U	VPI Research Reactor, Virginia	Thermal Spectrum	~0.04–319 s	5	LSF
^{233}U	East et al. (1970)	67, 74	~10 gms U metal 97.5% ^{233}U	Accelerator I, Los Alamos	14.7 MeV	0.02 – 385 s	6	LSF
^{233}U	Maksyutenko et al. (1971)	72	?	Van de Graaff, Titanium-Tritium target	18.0, 18.2, 18.5, 18.8, 19.0, 19.5, 20.0, 20.5, 21.0 MeV	5 – 512 s	11	LSF
^{233}U	Waldo et al. (1981)	82, 83	A few μg/mg of U: 95.1% ^{233}U 3.5% ^{238}U 0.8% ^{235}U 0.5% ^{239}U 0.1% ^{236}U 4 ppm ^{232}U	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ?s	6	LSF
^{233}U	Benedetti et al. (1982)	84	A few mg of oxide powder, 100% ^{233}U	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{233}U	Gudkov et al. (1989)	87	?	BR-1 Reactor, Obninsk, Russia	Fission Spectrum	0.8 – 600 s	6	LSF
^{235}U	Snell, et al. (1942, 1943)	17, 19	48.0 Kg of U_3O_8 natural enrichment	Univ. Chicago cyclotron: Be-target in paraffin	Thermal Spectrum	~0.5–780 s	4	Graphical
^{235}U	Wilson & Sutton (1944)	20,28	1.93 gms ^{235}U 15.5 gms ^{238}U in paraffin	Los Alamos cyclotron	Thermal Spectrum	~ few ms	?	?
^{235}U	Redman & Saxon (1944)	21, 31	1.28 gms “considerably enriched” U	Argonne Graphite Pile	Thermal Spectrum	0.6 – 1.0 s	4	Graphical
^{235}U	de Hoffmann et al. (1945)	23, 33	~73% enriched UH_{10}	Dragon Reactor, Los Alamos	~Thermal Spectrum	~0.2 s	5	Graphical
^{235}U	Hughes, Dabbs, et al. (1945)	24, 32	~3 gms U_3O_8 powder ~89% enriched	Argonne CP-3 Reactor	Thermal Spectrum	0.25 – 0.4 s	5	Graphical
^{235}U	Snell et al. (1946, 1947)	27, 29	Uranyl nitrate solution	Univ. Chicago cyclotron Be target	Thermal Spectrum	0.32 s	5	Graphical
^{235}U	Keepin et al. (1955)	36	A few gms ^{235}U isotopically pure	Godiva reactor, Los Alamos	~ Fission Spectrum	~0.05–300 s	6	LSF
^{235}U	Girshfeld (1955)	37	33.2 mg ^{235}U foil	? reactor	Thermal Spectrum	~0.4–180 s	5	Graphical

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{235}U	Rose & Smith (1957)	38	5.75 gms ^{235}U	ZEPHYR Reactor, Harwell	Fast Spectrum	~1–300 s	5	Graphical
^{235}U	Keepin et al. (1957)	39	~2–5 gms U: 99.9% ^{235}U	Godiva Reactor, LASL	Thermal & Fast Spectra	~0.05–330 s	6	LSF
^{235}U	Maksyutenko (1958)	41	~160 gms U	An accelerator—heavy ice target	Thermal, 2.4, 3.3, 15 MeV	~0.25–360 s	5	LSF
^{235}U	Hahn (1964)	46	9.15 gms U: 93% ^{235}U, 1% ^{234}U	Princeton Accelerator, USA	14 MeV	<0.48–400 s	4	LSF
^{235}U	Maksyutenko (1965)	49	U metal: 90% ^{235}U 10% ^{238}U	Van de Graff accelerator	6.0 MeV	~0.25–360 s	5	LSF
^{235}U	Maksyutenko (1967)	54	?	?	5.0, 6.0, 6.3, 6.6, 6.9, 7.22, 7.76 MeV	?	5	LSF
^{235}U	Maksyutenko et al. (1967)	55	?	?	18.5, 19.5, 20.0, 21.0 MeV	?	5	LSF
^{235}U	Schussler et al. (1968)	57	?	?	Thermal Spectrum	?	5	?
^{235}U	Cox & Whiting (1968)	59	~45 gms U	Van de Graaff, Li(p,n)	0.5, 0.6, 1.3 MeV	0.2 – 400 s	5	LSF
^{235}U	Rambo (1969)	63	15.0 gms U_3O_8 99.564% ^{235}U	VPI Research Reactor	Thermal Spectrum	~0.04–319 s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{235}U	Onega et al. (1969)	64	15 gms ^{235}U	VPI Research Reactor	Thermal Spectrum	~0.04–319 s	5	LSF
^{235}U	Auguston et al. (1969)	65, 69	~10 gms U metal: 93% ^{235}U	Cockroft–Walton, (D,T) target	14.9 MeV	0.02 – 303 s	6	LSF
^{235}U	East et al. (1970)	67, 74	~10 gms U metal 99% ^{235}U	Accelerator I, Los Alamos	14.7 MeV	0.02 – 385 s	6	LSF
^{235}U	Chrysochoides et al. (1970)	70	?	Democritos Reactor, Greece	Above Cd Cutoff	30 – 320 s	3	LSF
^{235}U	Schussler & Herrnann (1972)	76	?	?	Thermal Spectrum	?	6	LSF
^{235}U	Besant et al. (1977)	81	0.035 – 8.3 gm U metal: 95.54% ^{235}U 3.10% ^{238}U 1.20% ^{234}U 0.17% ^{236}U	VIPER Reactor, England	Fast Reactor Spectrum	0.035–900 s	6	LSF
^{235}U	Waldo et al. (1981)	82, 83	A few µg/mg of enriched uranium 93.7% ^{235}U	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ?s	5	LSF
^{235}U	Synetos & Williams (1983)	85	35 – 350 mg metal foils: 95.54% ^{235}U 4.46% ^{238}U	Univ. of London Reactor, England	Thermal Spectrum	0.38 – 380 s	5	LSF
^{235}U	Gudkov et al. (1989)	86	2.09 µg ^{235}U	IRT Reactor, Russia	Thermal Spectrum	~1 – ? s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{235}U	Gudkov et al. (1989)	88	?	BR-1 Reactor, Russia	Fast Spectrum	~1 – ? s	6	LSF
^{235}U	Charlton et al. (1996)	90, 93	12 mg U 97.663% ^{235}U 2.337% ^{238}U	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.51 – 900 s	5, 6	LSF
^{235}U	Saleh et al. (1997)	91	10 – 300 mg U: 97.663% ^{235}U	Texas A&M TRIGA Reactor, USA	Thermal Spectrum	0.44 – 350 s	5	LSF
^{235}U	Piksaikin et al. (1997)	92, 94, 97, 101	?	?	0.370, 0.624, 0.859, 1.059, 1.165 MeV	0.15 - 300 s	6, 8	LSF
^{235}U	Loaiza et al. (1997)	95	3 g U 93.5% ^{235}U 5.3% ^{238}U 1.2% ^{234}U	Godiva Reactor, LANL,USA	Fast Spectrum	0.075 - 300 s	6	LSF
^{235}U	Charlton et al. (1998)	98	12 mg U 97.663% ^{235}U 2.337% ^{238}U	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.51 – 900 s	6	LSF
^{235}U	Kellett. (1998)	99	40 gms U; 93% ^{235}U 7% ^{238}U	Dynamitron Accel., Birmingham, England	1.1–5.15 MeV	1.1 – 39.4 s	3, 6, 9, 12	LSF
^{236}U	Gudkov et al. (1989)	87	?	BR-1 Reactor, Russia	Fast Spectrum	0.8 – 600 a	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{238}U	Sun et al. (1950)	35	2.26 Kg ^{238}U metal: Cd covered 99.3% ^{238}U, 0.7% ^{235}U	Univ. of Pittsburgh cyclotron—C, LiF, B_4C targets	14, 29 MeV	? – 500 s	2–5	Graphical
^{238}U	Keepin et al. (1955)	36	A few gms ^{238}U isotopically pure	Godiva reactor, Los Alamos	~ Fission Spectrum	~0.05–300 s	6	LSF
^{238}U	Rose & Smith (1957)	38	72.5 gms ^{238}U: natural enrichment	ZEPHYR Reactor, Harwell	Fast Spectrum	~1 – 300 s	5	Graphical
^{238}U	Keepin et al. (1957)	39	~2–5 gms U: 99.98% ^{238}U	Godiva Reactor, Los Alamos	Fast Spectrum	~0.05–330 s	6	LSF
^{238}U	Maksyutenko (1958)	41	~160 gms U	An accelerator—heavy ice target	2.4, 3.3, 15 MeV	~0.25–360 s	5	LSF
^{238}U	Hermann et al. (1965)	47	A few gms ^{238}U: 99.7% ^{238}U 0.3% ^{235}U	An accelerator	14 MeV	?	4	?
^{238}U	Maksyutenko (1965)	48	?	Van de Graff $^{2}H(d,n)^{3}He$ $^{3}H(d,n)^{4}He$	1.5, 2.3, 3.8, 5.75, 6.5, 15 MeV	5.5 – ?s	5-6	LSF
^{238}U	Maksyutenko (1965)	49	^{238}U metal thick target 20 mg/cm^2	Van de Graff accelerator	1.75 MeV max	~0.25–360 s	5	LSF
^{238}U	Bucko (1966)	51	natural enrichment U_3O_8	Atomki Neutron Generator	14.7 MeV	0.5 – ? s	5	Graphical

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{238}U	Hermann (1967)	53	Several gms U metal: 99.7% ^{238}U, 0.3% ^{235}U	An accelerator–Be-D reaction	14MeV	0.3 – 400 s	6	LSF
^{238}U	Maksyutenko (1967)	54	?	Van de Graaff	5.0, 6.0, 6.4, 6.6, 6.8, 6.9, 7.1, 7.25, 7.5, 7.76 MeV	?	5	LSF
^{238}U	Maksyutenko et al. (1967)	55	?	Van de Graaff	18.0, 19.0, 19.5, 20.0 MeV	?	5	LSF
^{238}U	Maksyutenko et al. (1968)	58	?	Van de Graaff, Tritium-Zr target	18.2, 18.5, 18.8, 19.3, 19.7, 20.5, 21.0 MeV	?	5	LSF
^{238}U	Cox & Whiting (1968)	59	~45 gms U	Van de Graaff, Li(p,n)	1.4, 1.5, 1.75 MeV	0.2 – 400 s	5	LSF
^{238}U	Notea (1969)	62	?	?	14 MeV	?	5	?
^{238}U	Auguston et al. (1969)	65, 69	10 gms U metal: 99.7% ^{238}U	Cockroft–Walton, (D,T) target	14.9 MeV	0.02 – 303 s	6	LSF
^{238}U	East et al. (1970)	67, 74	~10 gms U metal: 99.6% ^{238}U	Accelerator I, Los Alamos	14.7 MeV	0.02 – 385 s	6	LSF
^{238}U	Brown et al. (1971)	71	~10 gms depleted UO_3	S.A.M.E.S. neutron generator, $^{3}H(d,n)^{4}He$	14.8 MeV	?	4	?
^{238}U	Benedict et al. (1972)	75	?	?	14.8 MeV	?	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{238}U	Maksyutenko et al. (1974)	80	30 gm U metal	KG-2.5 Accelerator, Titanium-deuterium target	3.9, 4.2, 4.5, 4.8, 5.1 MeV	5 – 1024 s	4	LSF
^{238}U	Besant et al. (1977)	81	0.25 – 11.2 gm U metal: 99.61% ^{238}U 0.39% ^{235}U	VIPER Reactor, England	Fast Reactor Spectrum	0.035 - 900 s	6	LSF
^{238}U	Waldo et al. (1981)	82, 83	A few μg/mg of ultra-pure uranium 99.999% ^{238}U	Livermore Pool-Type, Thermal Reactor, LLNL	Fast Component of Thermal Spectrum	0.3 – ?s	6	LSF
^{238}U	Kellett. (1998)	99	9.6, 24, 48 gms U; 99.7% ^{238}U 0.3% ^{235}U	Dynamitron Accel., Birmingham, England	1.1–5.15 MeV	1.1 – 39.4 s	3, 6, 9, 12	LSF
^{237}Np	Sanguist et al. (1973)	78	30 mg of Np: ~2% ^{228}Th	L54 Reactor	Fast Spectrum	1 – ? s	2	LSF
^{237}Np	Maksyutenko et al. (1974)	79	?	Van de Graaff, T(p,n)^{3}He	0.4, 0.5, 0.6, 0.7, 0.8, 0.9, 1.0, 1.1, 1.2 MeV	5 – 1024 s	4	LSF
^{237}Np	Waldo et al. (1981)	82, 83	A few μg/mg of Np: 99.19% ^{237}Np 0.7% Th 0.1% U 0.01% Pu	Livermore Pool-Type, Thermal Reactor, LLNL	Fast Component of Reactor Spectrum	0.3 – ? s	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{237}Np	Benedetti et al. (1982)	84	A few mg of oxide powder 100% ^{237}Np	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF
^{237}Np	Gudkov et al. (1989)	87	?	BR-1 Reactor, Russia	Fast Spectrum	0.8 – 600 a	6	LSF
^{237}Np	Saleh et al. (1995)	89, 91	10 mg Np 99.999% ^{237}Np	Texas A&M Reactor, USA	20% Thermal, 80% Fast Spectrum	0.44 – 350 s	5	LSF
^{237}Np	Charlton et al. (1996)	90, 93	10 mg Np 99.999% ^{237}Np	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.51 – 900 s	6	LSF
^{237}Np	Loaiza. (1997)	96	4 g ^{237}Np 99.19% 0.81 other	Godiva Los Alamos, USA	Fast Spectrum	0.075 - 300 s	6	LSF
^{237}Np	Piksaikin et al. (1997)	92, 94, 97, 101	?	?	0.586, 1.008, 3.745, 4.196, 4.719 MeV	0.15 - 300 s	6, 8	LSF
^{237}Np	Charlton et al. (1998)	98, 100	10 mg Np 99.999% ^{237}Np	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.5 – 900 s	6, 7	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{238}Pu	Waldo et al. (1981)	82, 83	A few μg/mg of isotopically purified Pu 99.8% ^{238}Pu 0.1% ^{239}Pu <0.1% ^{238}U	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ? s	6	LSF
^{238}Pu	Benedetti et al. (1982)	84	A few mg of oxide powder 92.43% ^{238}Pu 4.88% ^{234}U, 1.10% ^{237}Np, 0.64% ^{239}Pu 0.89% ^{240}Pu, 0.04% ^{241}Am 0.02% ^{242}Pu	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF
^{239}Pu	Wilson & Sutton (1944)	20,28	0.565 gms ^{239}Pu surrounded by paraffin	Los Alamos cyclotron, USA	Thermal Spectrum	~ few ms	—	—
^{239}Pu	Redman & Saxon (1944)	21, 31	1.0862 gms PuO_2	Graphite Pile, ANL, Chicago, USA	Thermal Spectrum	0.6 – 1.0 s	4	Graphical
^{239}Pu	Feld & de Hoffmann (1945)	22	5.6 gms ^{239}Pu metal with Cd cover (purity not reported)	Water Boiler Reactor, Los Alamos, USA	~Thermal Spectrum	2 or 5 s	4	Graphical
^{239}Pu	Perry et al. (1946)	26	118 gms of Pu	electrostatic gen.— ^{7}Li(p,n)^{7}Be	0.4–0.6 MeV	0.01 s	5	Graphical
^{239}Pu	de Hoffmann & Feld (1947)	30	^{239}Pu metal with Cd cover	Water Boiler Reactor, Los Alamos, USA	~Thermal Spectrum	2	4	Graphical

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{239}Pu	Keepin et al. (1955)	36	A few gms ^{239}Pu	Godiva reactor, Los Alamos, USA	~ Fission Spectrum	~0.05–300 s	6	LSF
^{239}Pu	Rose & Smith (1957)	38	9.74 gms ^{239}Pu	ZEPHYR Reactor, Harwell, England	Fast Spectrum	~1 – 300 s	5	Graphical
^{239}Pu	Keepin et al. (1957)	39	~2–5 gms Pu 99.8% ^{239}Pu	Godiva Reactor, Los Alamos, USA	Thermal & Fast Spectra	~0.05–330 s	6	LSF
^{239}Pu	Maksyutenko (1963)	44	?	? accelerator, Russia	3.8, 15 MeV	? – 280 s	5, 6	LSF
^{239}Pu	Maksyutenko (1967)	54	?	?	5.5, 6.5, 7.0, 7.5, 7.8 MeV	?	5	LSF
^{239}Pu	Huizinga (1968)	60	17.01 gms Plutonium: 0.030% ^{238}Pu 94.466% ^{239}Pu 4.762% ^{240}Pu 0.508% ^{241}Pu 0.270% ^{242}Pu	VPI Research Reactor, Virginia, USA	Thermal Spectrum	~0.04–300 s	5	LSF
^{239}Pu	Onega et al. (1969)	64	15 gms ^{239}Pu	VPI Research Reactor, Virginia, USA	Thermal Spectrum	0.04–319 s	5	LSF
^{239}Pu	Maksyutenko et al. (1971)	72	?	Van de Graaff, Titanium-Tritium target	18.0, 18.2, 18.5, 18.8, 19.0, 19.5, 20.0, 20.5, 21.0 MeV	5 – 512 s	11	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{239}Pu	Besant et al. (1977)	81	0.18 – 12.39 gm Pu metal 93.85% ^{239}Pu 5.79% ^{240}Pu 0.36% ^{241}Pu	VIPER Reactor, England	Fast Reactor Spectrum	0.035 –900 s	6	LSF
^{239}Pu	Waldo et al. (1981)	82, 83	A few μg/mg of chemically purified plutonium 93.6% ^{239}Pu 5.7% ^{240}Pu 0.65% ^{241}Pu 0.01% ^{238}Pu	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ? s	6	LSF
^{240}Pu	Keepin et al. (1957)	39	~2–5 gms Pu 81.5% ^{240}Pu	Godiva Reactor, Los Alamos, USA	Fast Spectrum	~0.05–330 s	6	LSF
^{240}Pu	Benedetti et al. (1982)	84	A few mg of oxide powder 98.07% ^{240}Pu, 0.90% ^{239}Pu, 0.53% ^{241}Pu, 0.29% ^{242}Pu, 0.21% ^{241}Am,	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF
^{240}Pu	Gudkov et al. (1989)	87	?	BR-1 Reactor, Russia	Fast Spectrum	0.8 – 600 a	6	LSF
^{241}Pu	Cox (1961)	43	2.55 mg ^{241}Pu as film on a Platinum disk	Argonne Research reactor	Thermal Spectrum	0.5 – 1000 s	5	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{241}Pu	Waldo et al. (1981)	82, 83	A few µg/mg of isotopically separated plutonium 99.8% ^{241}Pu <0.1% ^{240}Pu 0.1% ^{242}Pu	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ?s	6	LSF
^{241}Pu	Benedetti et al. (1982)	84	A few mg of oxide powder 81.94% ^{241}Pu, 17.66% ^{241}Am, 0.12% ^{237}Np 0.07% ^{239}Pu, 0.13% ^{240}Pu, 0.08% ^{242}Pu	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF
^{241}Pu	Gudkov et al. (1989)	87	?	BR-1 Reactor, Obninsk, Russia	Fast Spectrum	0.8 – 600 a	6	LSF
^{242}Pu	Auguston et al. (1969)	66	A few gms Pu 99.91% ^{242}Pu	Accelerator I, Los Alamos, USA	14.7 MeV	~0.02–302 s	6	LSF
^{242}Pu	East et al. (1970)	68	A few gms Pu 99.91% ^{242}Pu	Accelerator I, Los Alamos, USA	14.7 MeV	~0.02–302 s	6	LSF
^{242}Pu	Waldo et al. (1981)	82, 83	A few µg/mg of high purity plutonium 99.90% ^{242}Pu	Livermore Pool-Type, Thermal Reactor, LLNL	Fast Component of Reactor Spectrum	0.3 – ? s	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{241}Am	Sanguist et al. (1973)	78	5 mg of Am <1000 ppm U 700 ppm Pu	L54 Reactor, Italy	Fast Spectrum	1 – ? s	2	LSF
^{241}Am	Waldo et al. (1981)	82, 83	Chemically purified americium 98.1% ^{241}Am 1.8% ^{237}Np <0.1% other	Livermore Pool-Type, Thermal Reactor, LLNL	Thermal Spectrum	0.3 – ?s	5	LSF
^{241}Am	Benedetti et al. (1982)	84	A few mg of oxide powder 97.37% ^{241}Am, 1.88% ^{237}Np 0.75% ^{239}Pu	L54 reactor, Italy	Fast Spectrum	0.6 – 700 s	5	LSF
^{241}Am	Gudkov et al. (1989)	88	?	BR-1 Reactor, Obninsk, Russia	Fast Spectrum	~1 – ? s	6	LSF
^{241}Am	Saleh et al. (1995)	89, 91	? mg 100% ^{241}Am	Texas A&M TRIGA Reactor, USA	Thermal Spectrum	0.44 – 350 s	5	LSF
^{242}Am^{m}	Waldo et al. (1981)	82, 83	Isotopically purified americium 99.21% ^{242}Am^{m} 0.79% ^{241}Am <0.007% ^{243}Am	Livermore Pool-Type, Thermal Reactor LLNL	Thermal Spectrum	0.3 – ? s	6	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{243}Am	Saleh et al. (1995)	89, 91	10 – 25 mg 99.997% ^{243}Am	Texas A&M TRIGA Reactor, USA	Predominatedly Fast Spectrum	0.44 – 350 s	5	LSF
^{243}Am	Charlton et al. (1997)	93	10 – 25 mg Am 99.987% ^{243}Am	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.51 – 900 s	6	LSF
^{243}Am	Charlton et al. (1998)	98, 100	10 – 25 mg Am 99.987% ^{243}Am	Texas A&M TRIGA Reactor, USA	Fast Spectrum	0.51 – 900 s	6, 7	LSF
^{245}Cm	Waldo et al. (1981)	82, 83	Isotopically purified curium 99.26% ^{245}Cm 0.281% ^{244}Cm 0.215% ^{246}Cm 0.013% ^{247}Cm	Livermore Pool-Type, Thermal Reactor LLNL	Thermal Spectrum	0.3 – ? s	6	LSF
^{249}Cf	Waldo et al. (1981)	82, 83	Chemically purified californium 99.9% ^{249}Cf <0.1% other	Livermore Pool-Type, Thermal Reactor LLNL	Thermal Spectrum	0.3 – ? s	4	LSF
^{252}Cf	Smith et al. (1958)	40, 42	100% ^{252}Cf	^{252}Cf source	Spontaneous Fission	~0.3 – ? s	3	LSF

Table III. Summary of Experimental Results.

Isotope	Author(s)	Ref.	Sample Composition	Neutron Source	Incident Neutron Energy	Counting Interval	No. of Groups	Analysis Method
^{252}Cf	Chulick et al. (1969)	61, 73	10 μg ^{252}Cf, covered with gold foil	^{252}Cf source	Spontaneous Fission	0.7 – s	4	LSF

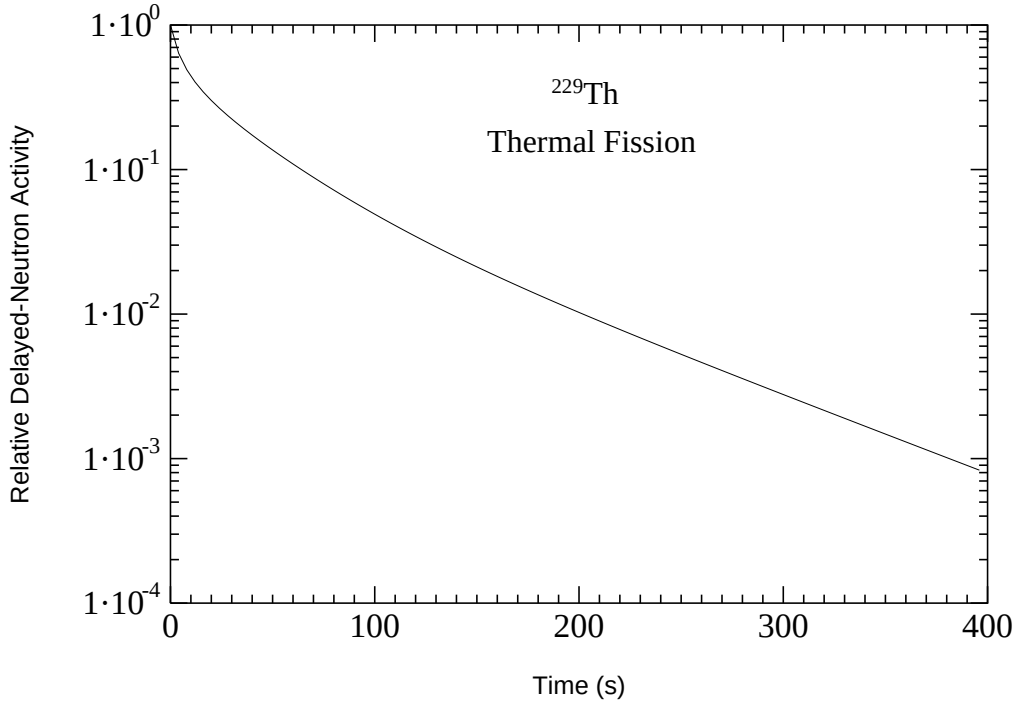

Fig. 5. Delayed Neutron Decay Curve for Thermal Fission of ^{229}Th.

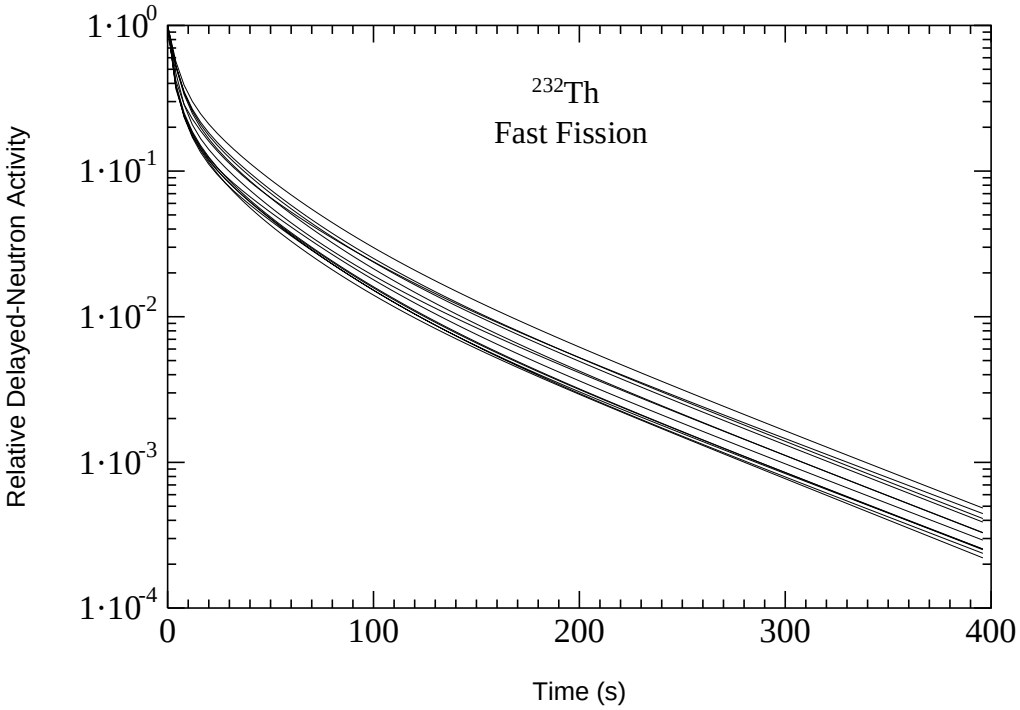

Fig. 6. Delayed Neutron Decay Curve for Fast Fission of ^{232}Th.

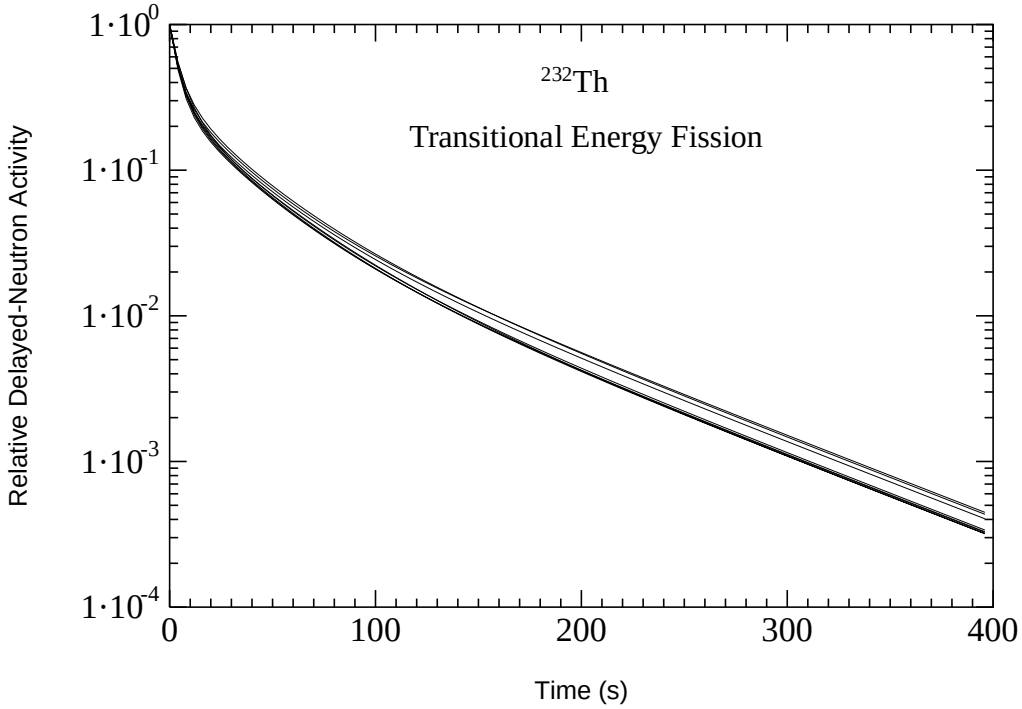

Fig. 7. Delayed Neutron Decay Curve for Transitional Energy Fission of ^{232}Th.

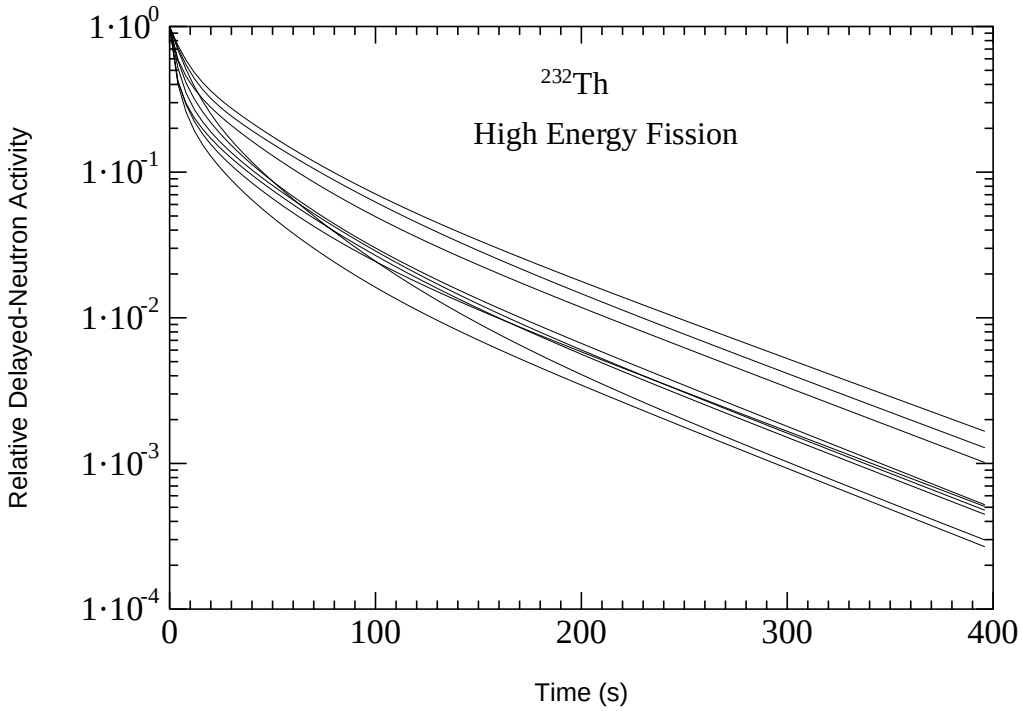

Fig. 8. Delayed Neutron Decay Curve for High Energy Fission of ^{232}Th.

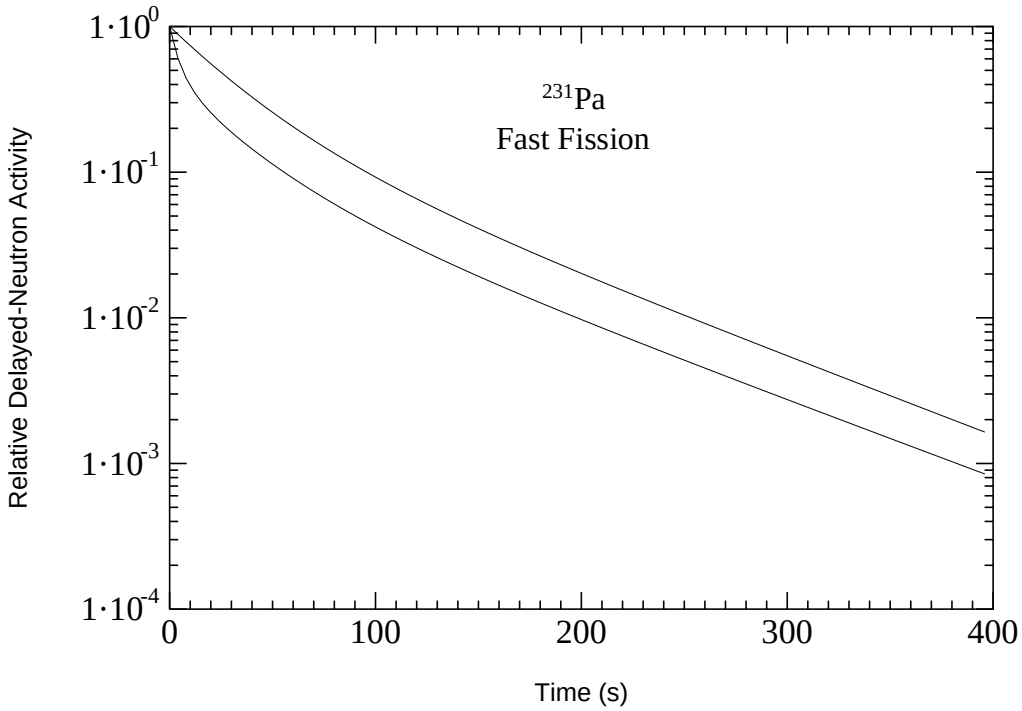

Fig. 9. Delayed Neutron Decay Curve for Fast Fission of ^{231}Pa.

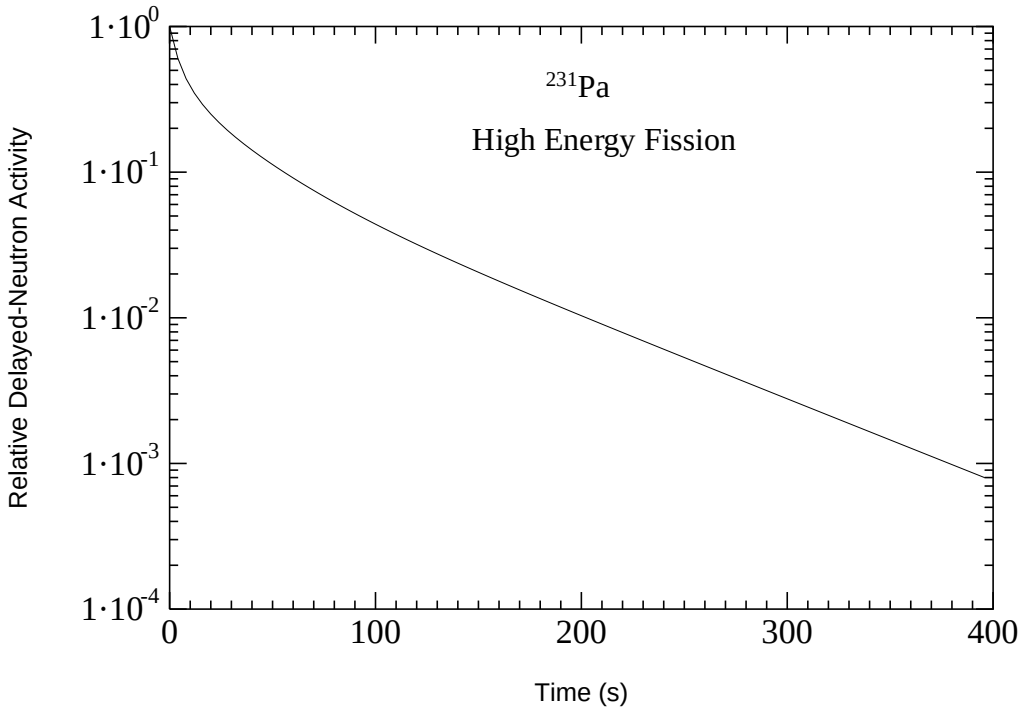

Fig. 10. Delayed Neutron Decay Curve for High Energy Fission of ^{231}Pa.

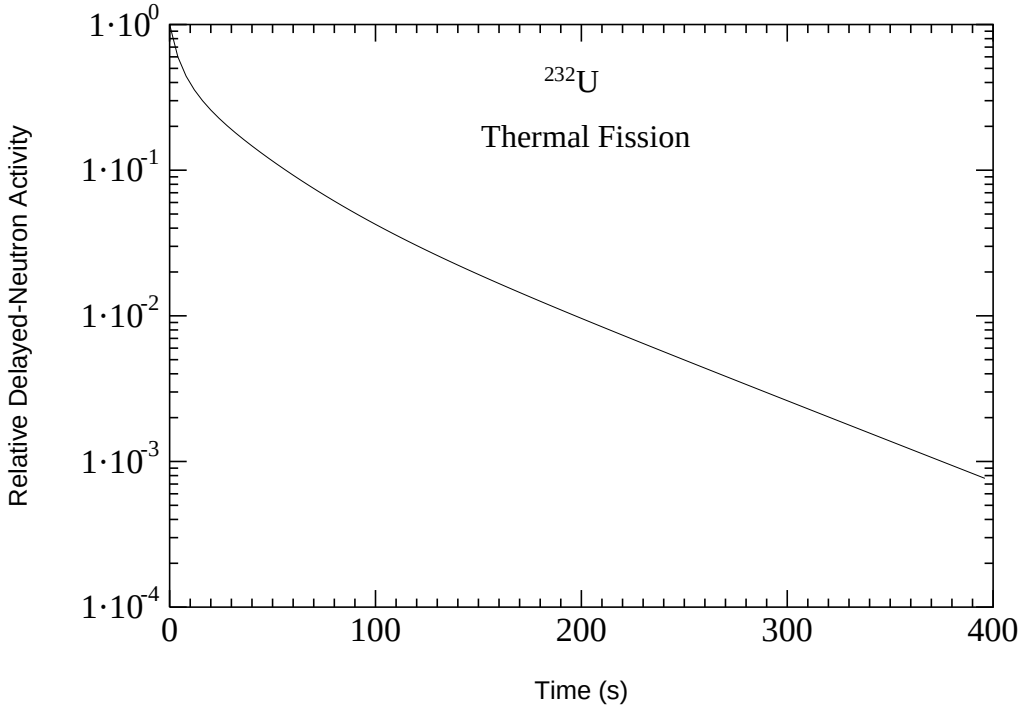

Fig. 11. Delayed Neutron Decay Curve for Thermal Fission of ^{232}U.

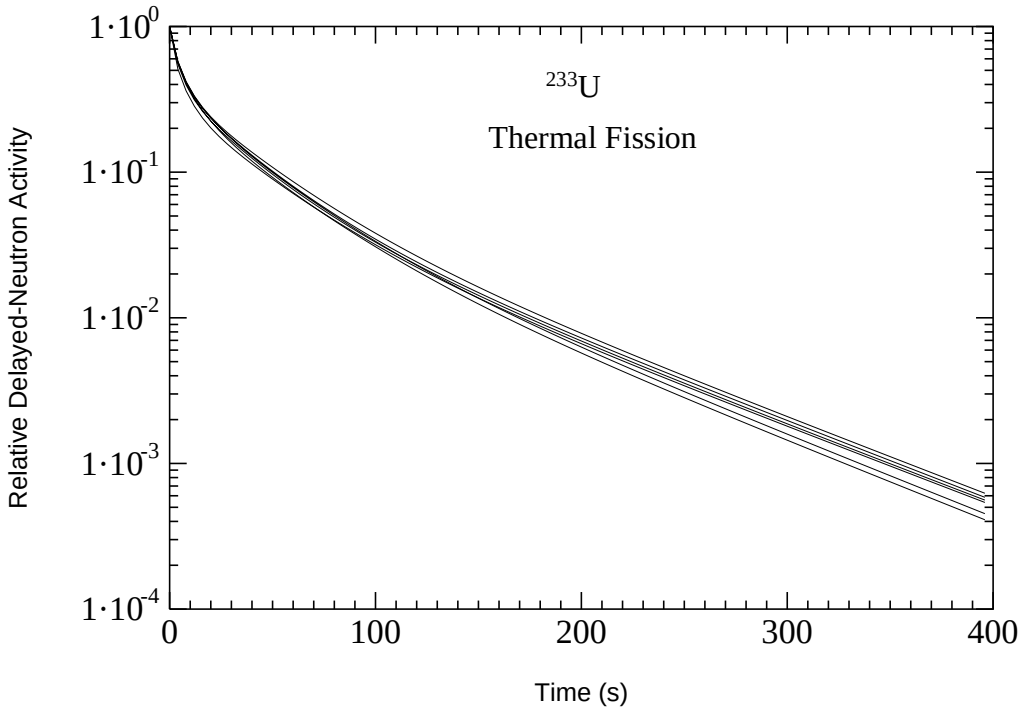

Fig. 12. Delayed Neutron Decay Curve for Thermal Fission of ^{233}U.

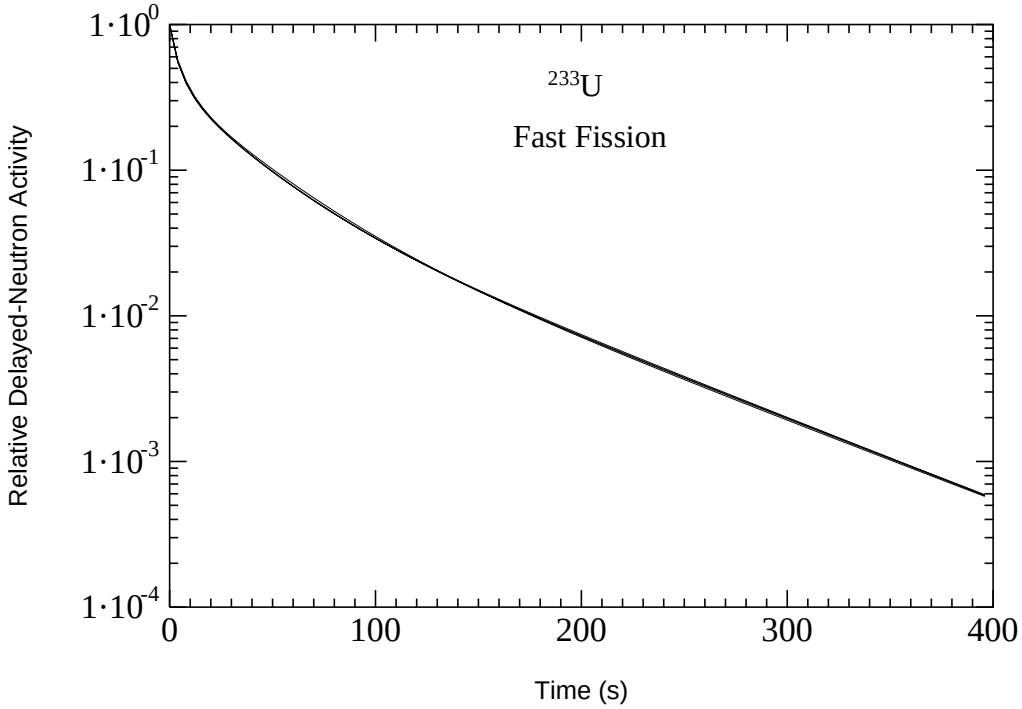

Fig. 13. Delayed Neutron Decay Curve for Fast Fission of ^{233}U.

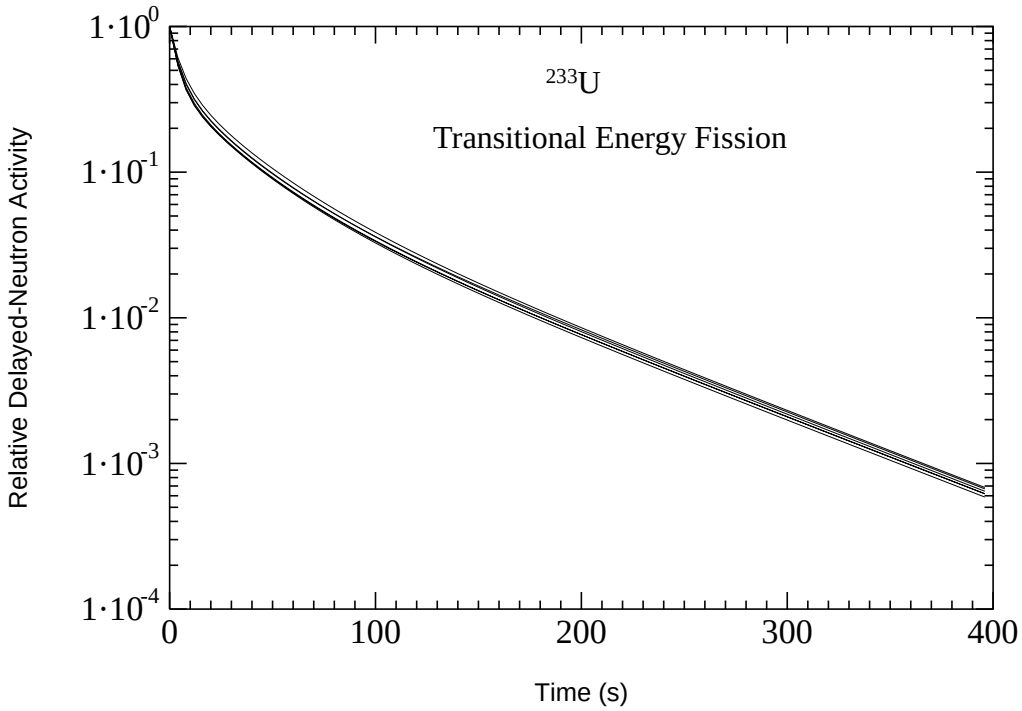

Fig. 14. Delayed Neutron Decay Curve for Transitional Energy Fission of ^{233}U.

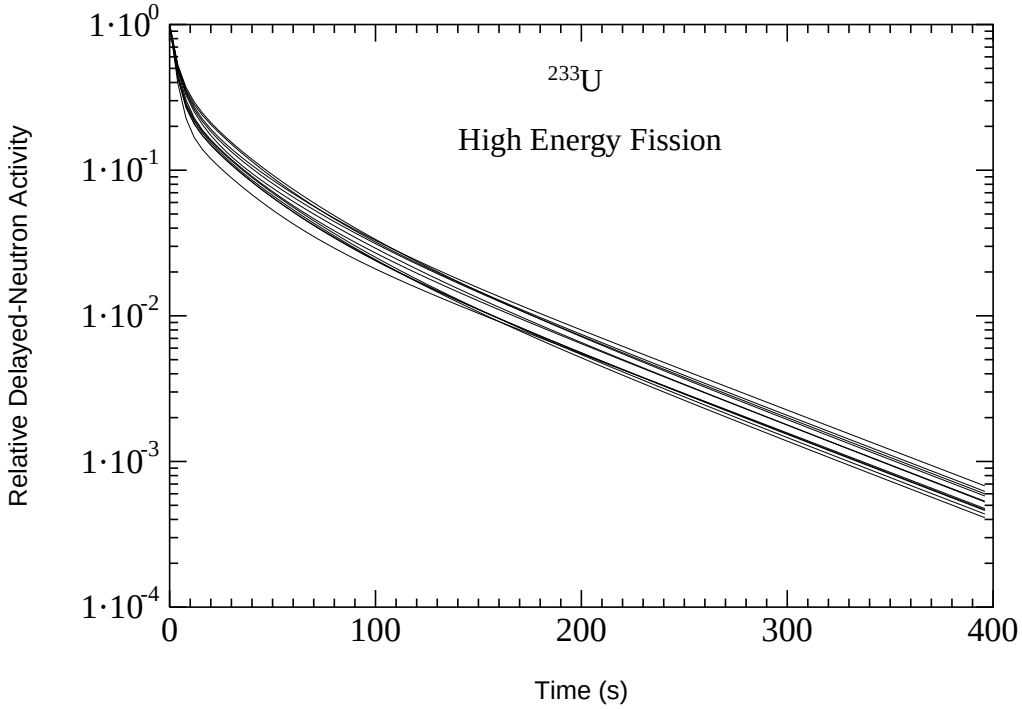

Fig. 15. Delayed Neutron Decay Curve for High Energy Fission of ^{233}U.

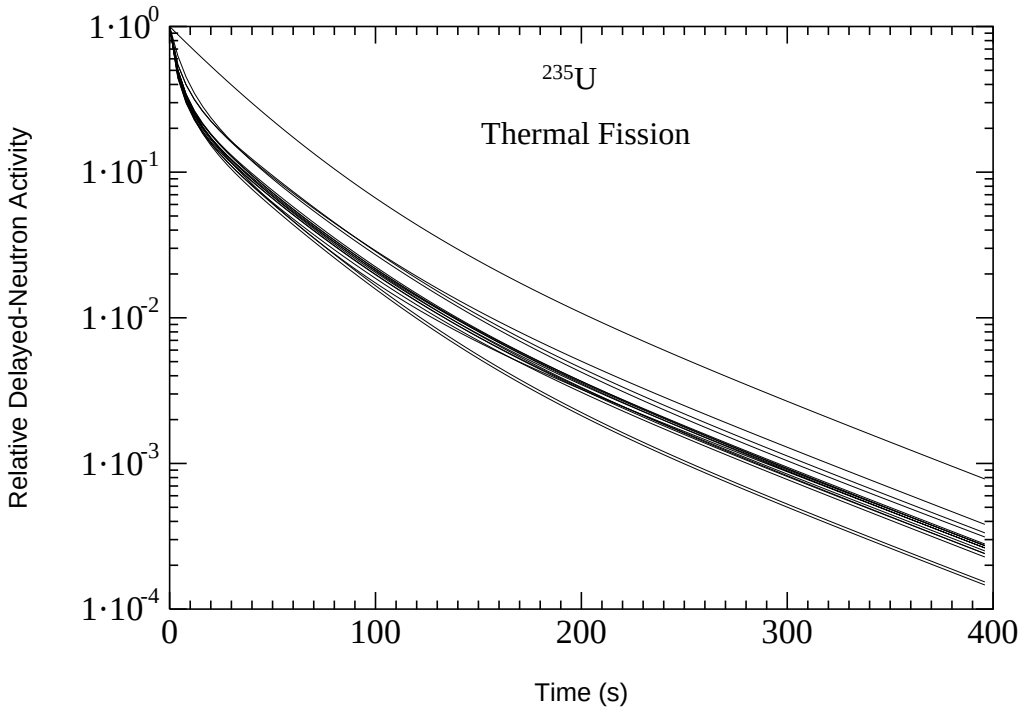

Fig. 16. Delayed Neutron Decay Curve for Thermal Fission of ^{235}U.

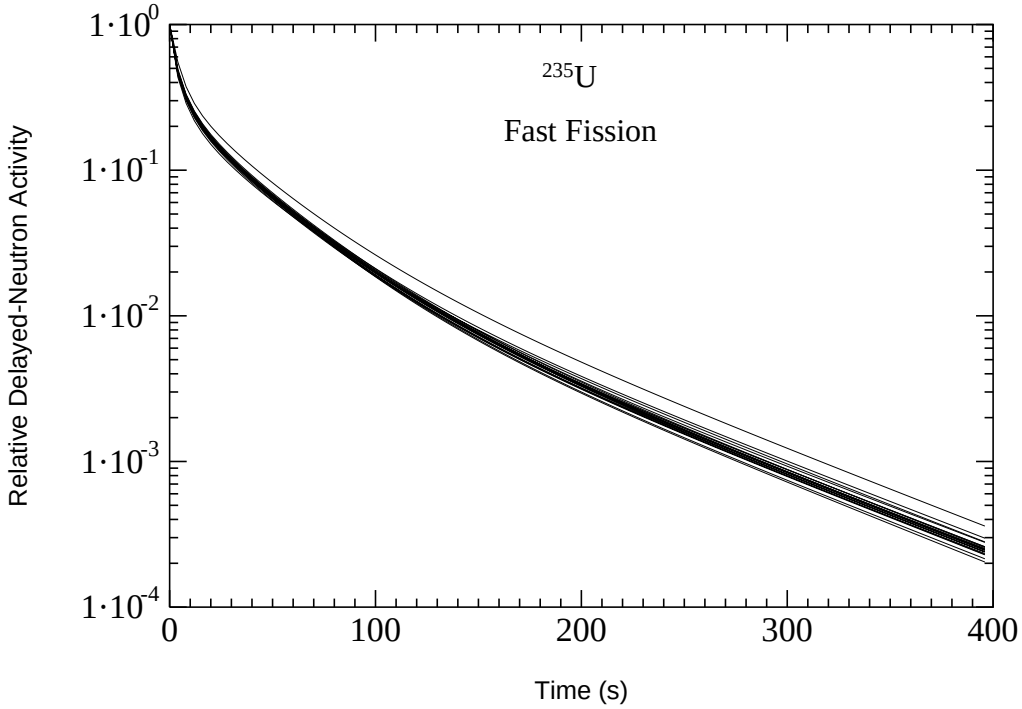

Fig. 17. Delayed Neutron Decay Curve for Fast Fission of ^{235}U.

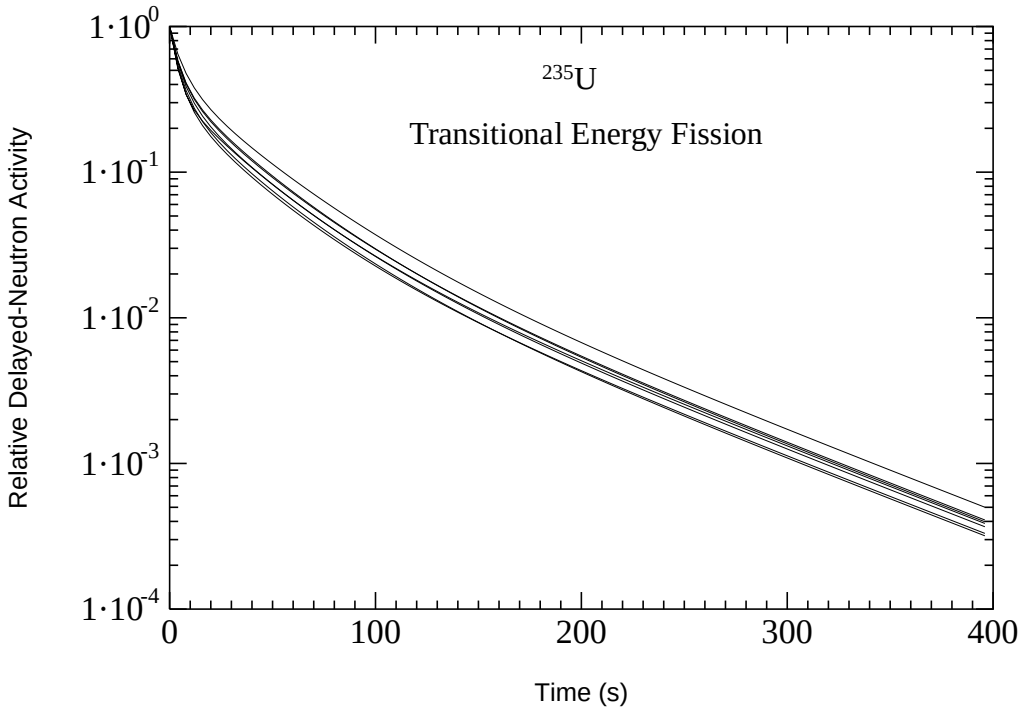

Fig. 18. Delayed Neutron Decay Curve for Transitional Energy Fission of ^{235}U.

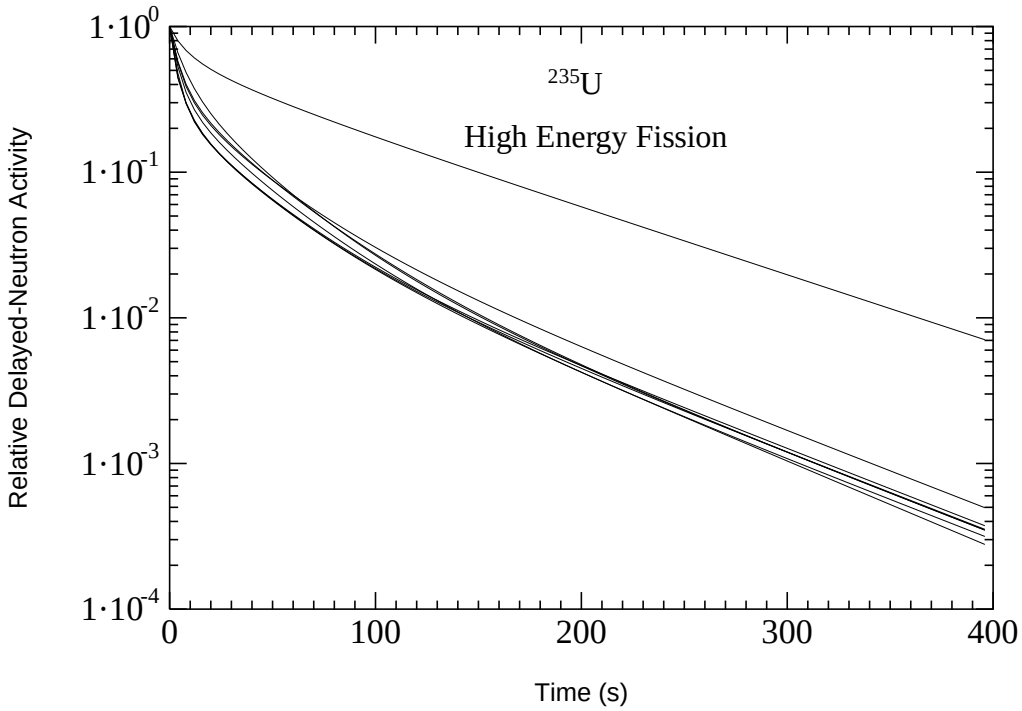

Fig. 19. Delayed Neutron Decay Curve for High Energy Fission of ^{235}U.

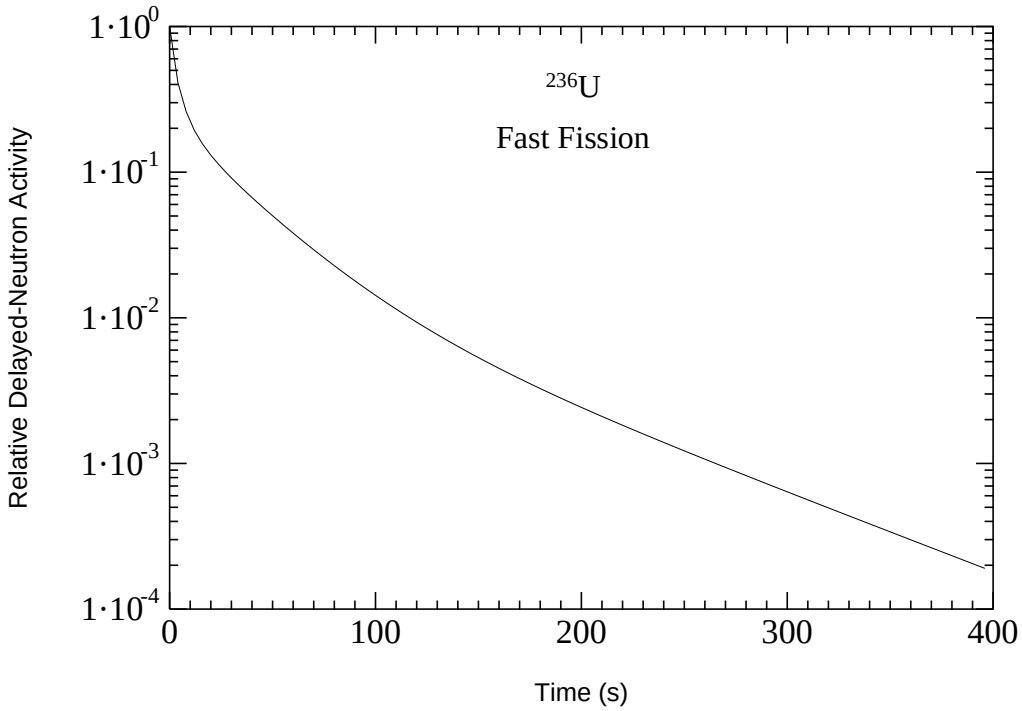

Fig. 20. Delayed Neutron Decay Curve for Fast Fission of ^{236}U.

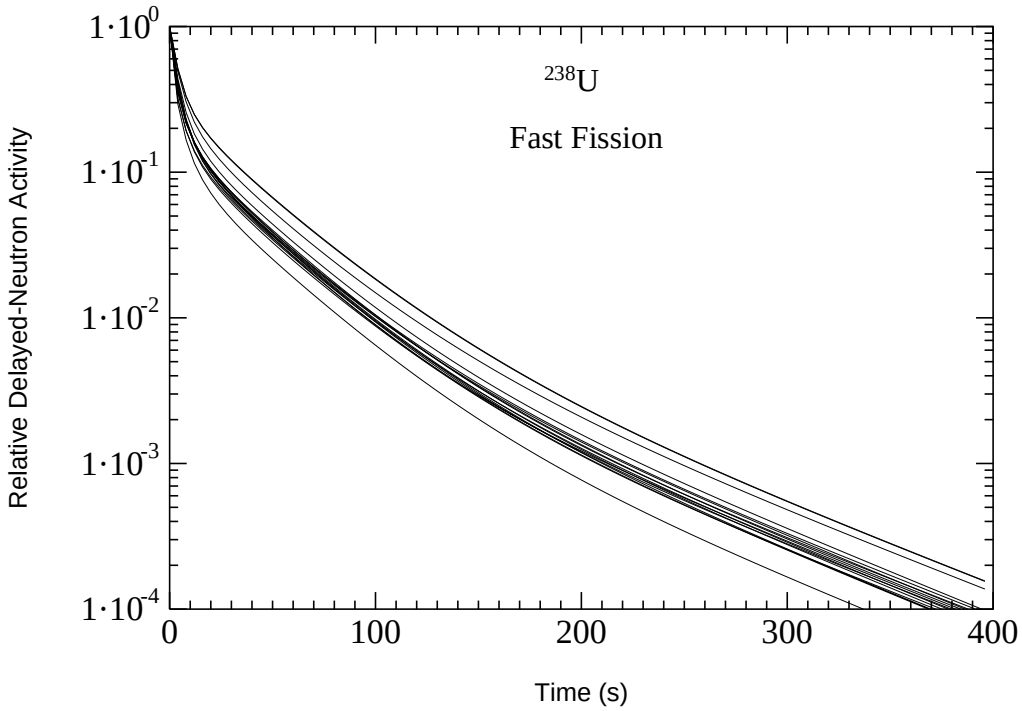

Fig. 21. Delayed Neutron Decay Curve for Fast Fission of ^{238}U.

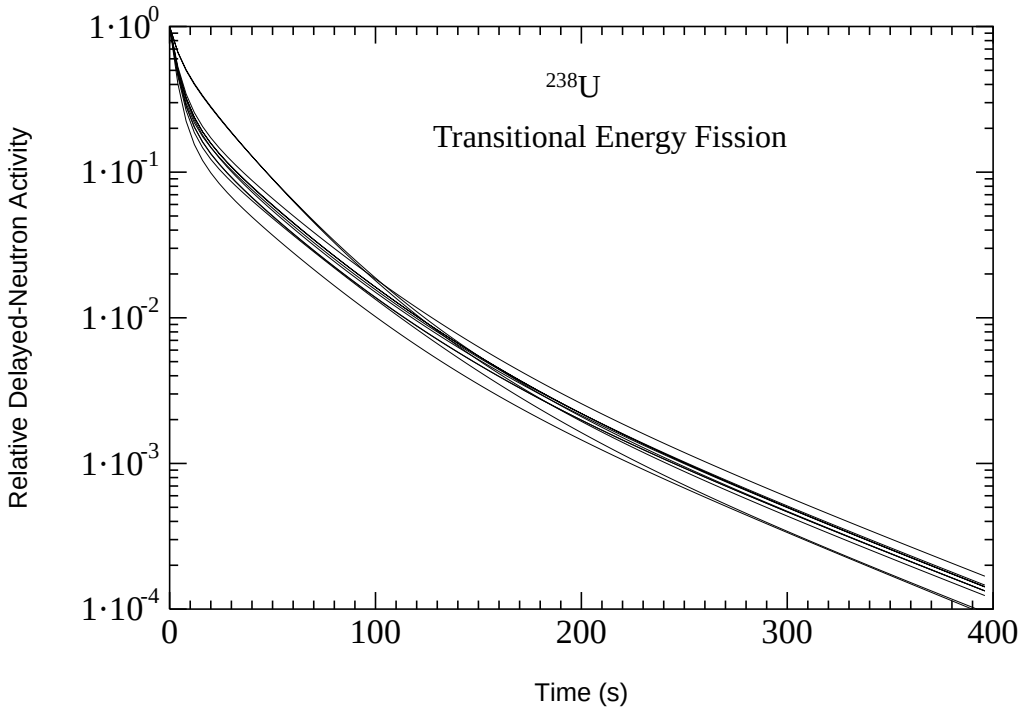

Fig. 22. Delayed Neutron Decay Curve for Transitional Energy Fission of ^{238}U.

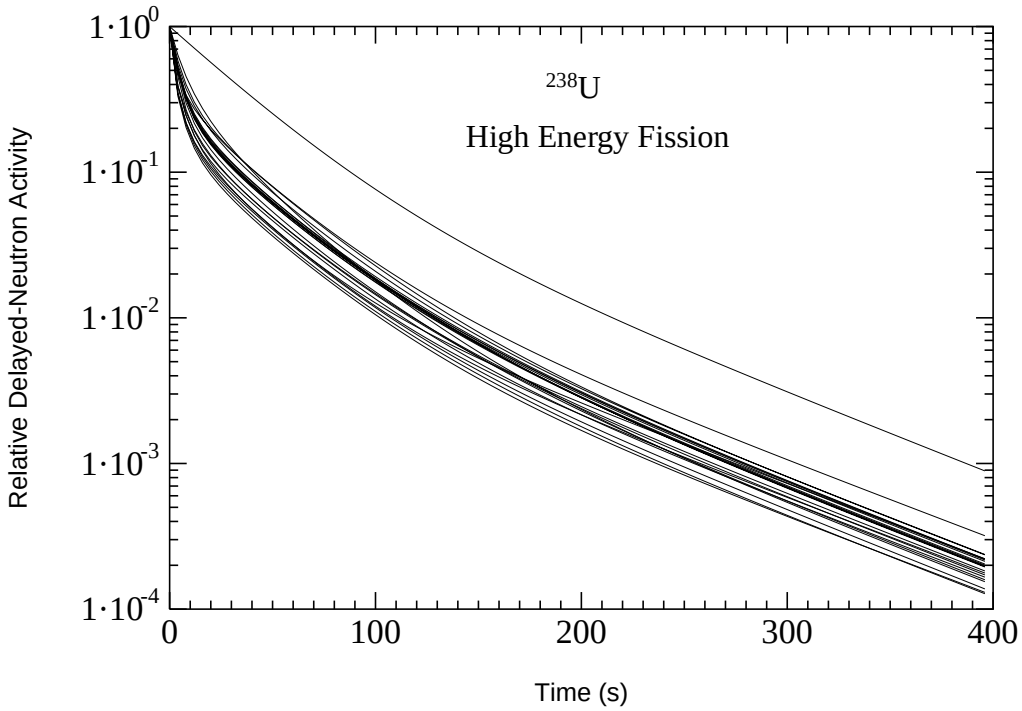

Fig. 23. Delayed Neutron Decay Curve for High Energy Fission of ^{238}U.

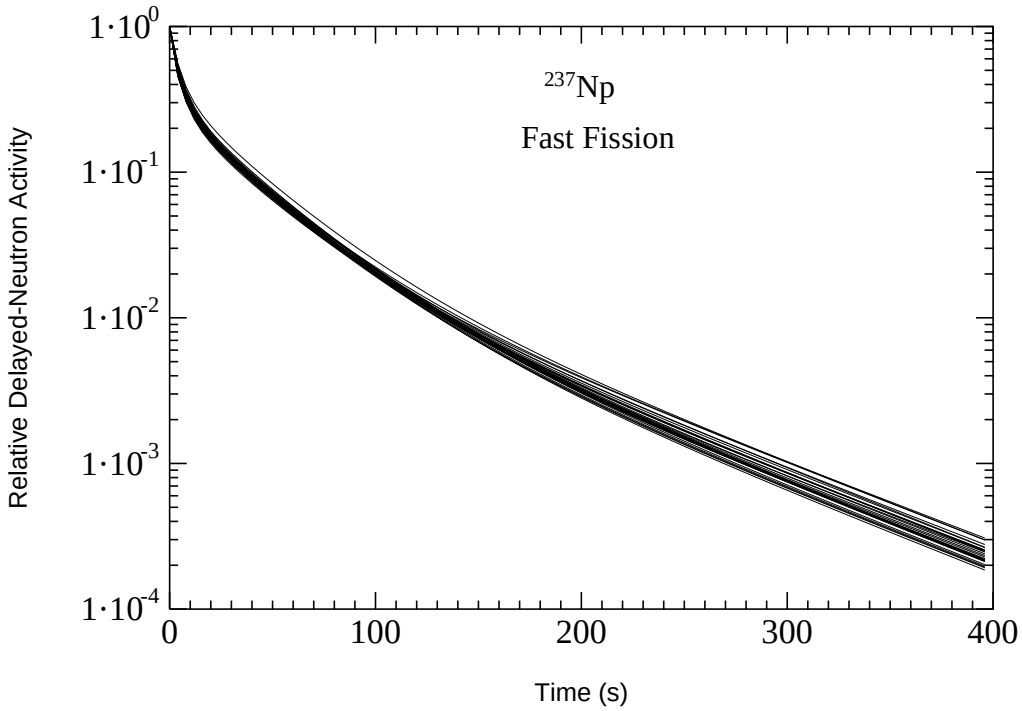

Fig. 24. Delayed Neutron Decay Curve for Fast Fission of ^{237}Np.

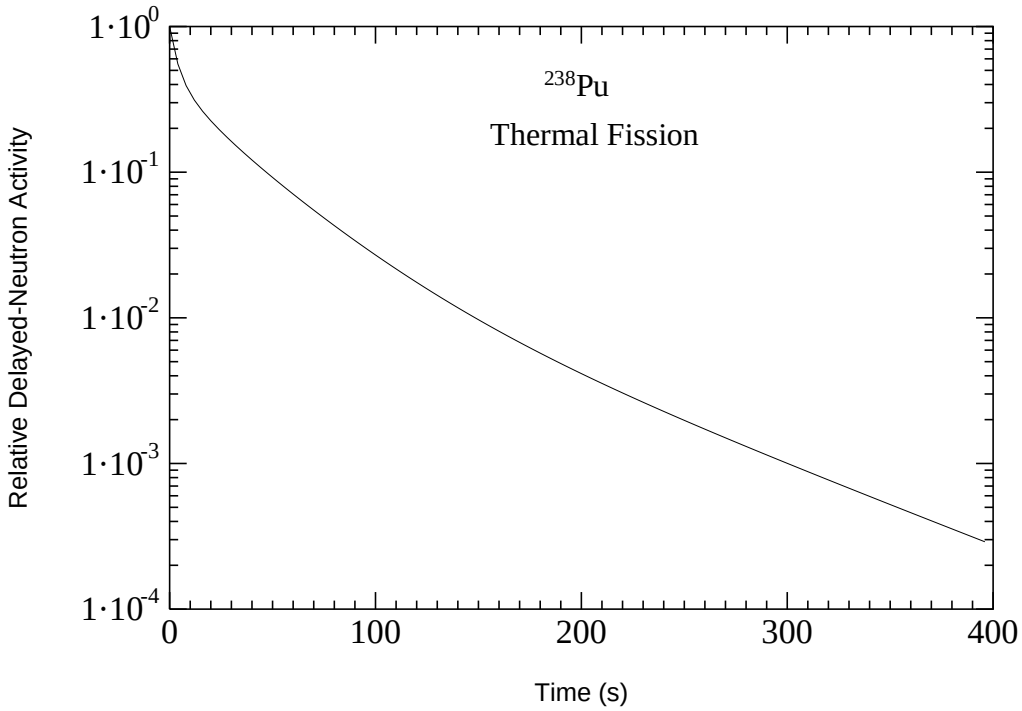

Fig. 25. Delayed Neutron Decay Curve for Thermal Fission of ^{238}Pu.

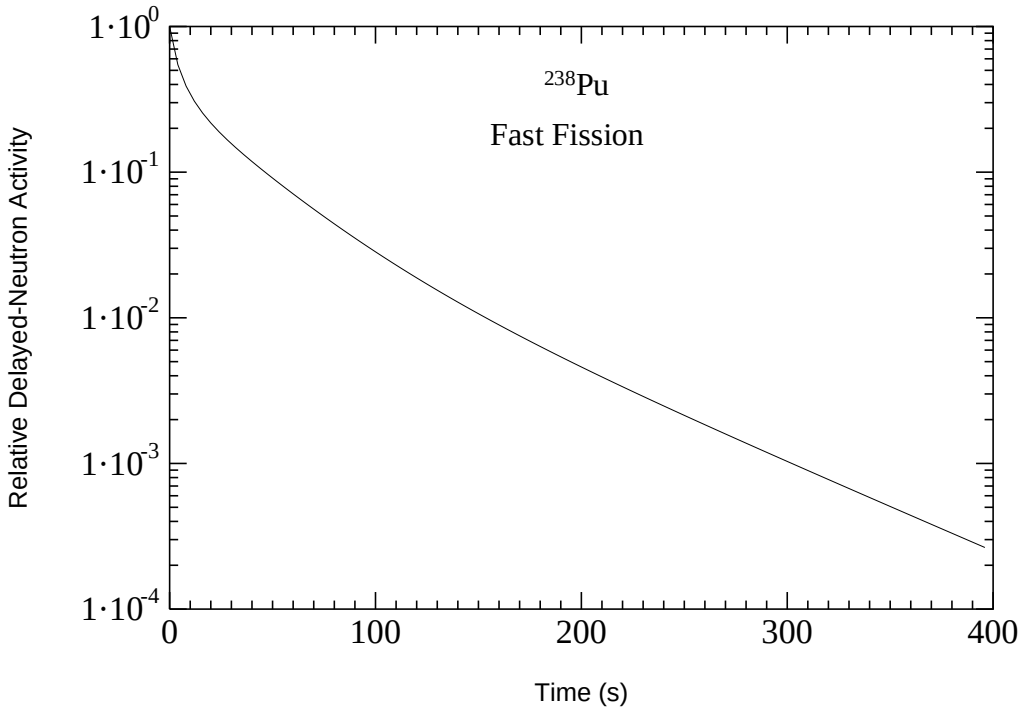

Fig. 26. Delayed Neutron Decay Curve for Fast Fission of ^{238}Pu.

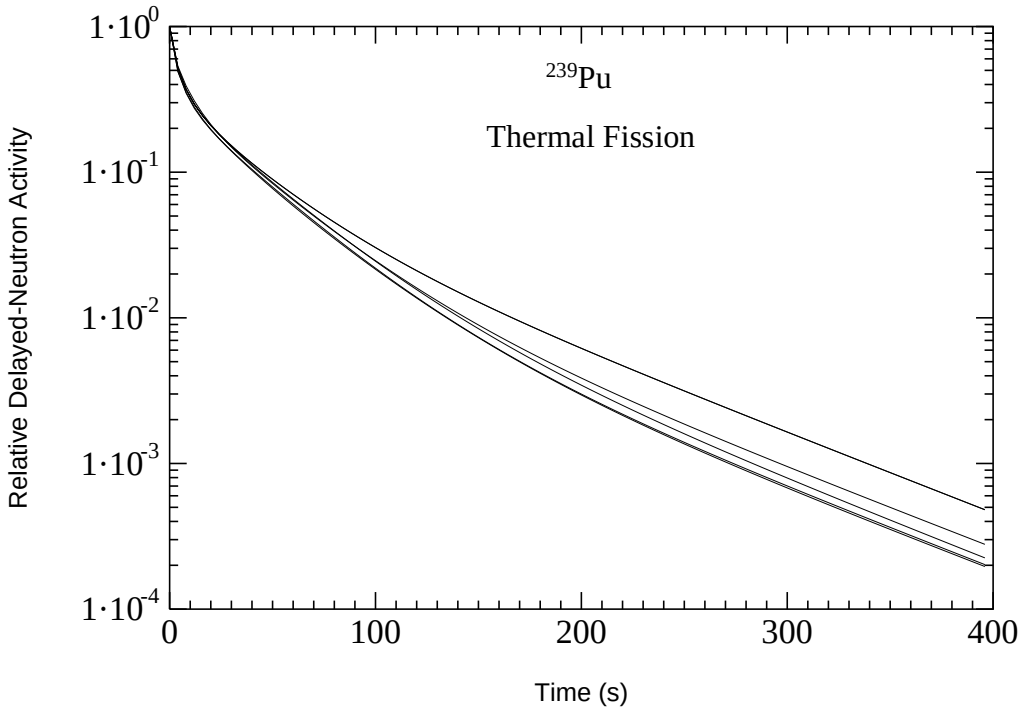

Fig. 27. Delayed Neutron Decay Curve for Thermal Fission of ^{239}Pu.

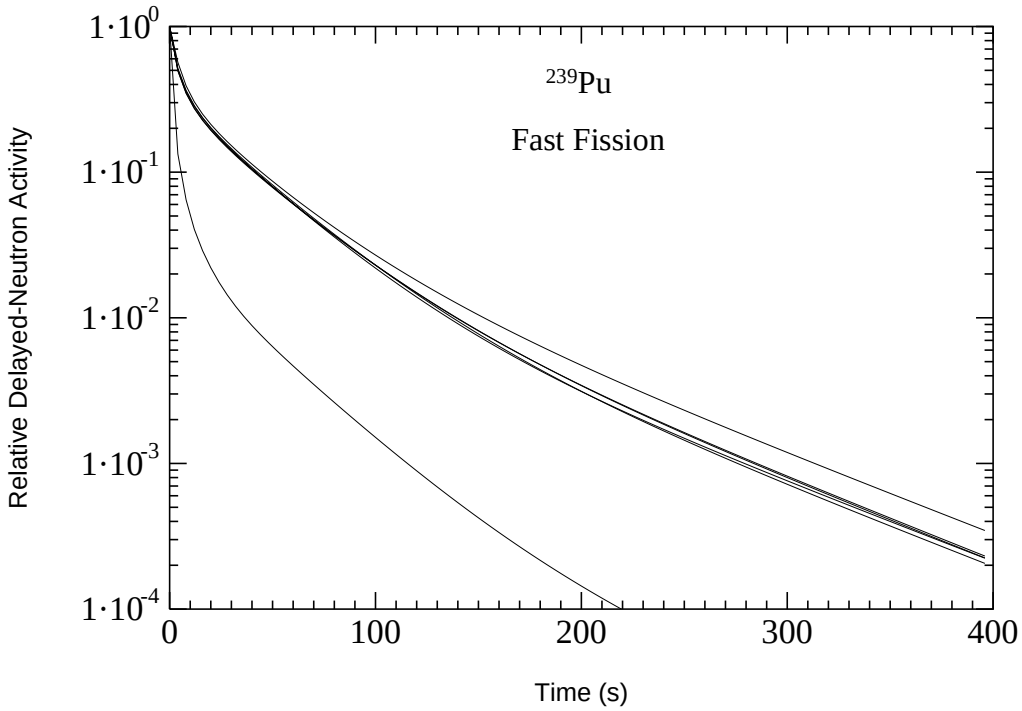

Fig. 28. Delayed Neutron Decay Curve for Fast Fission of ^{239}Pu.

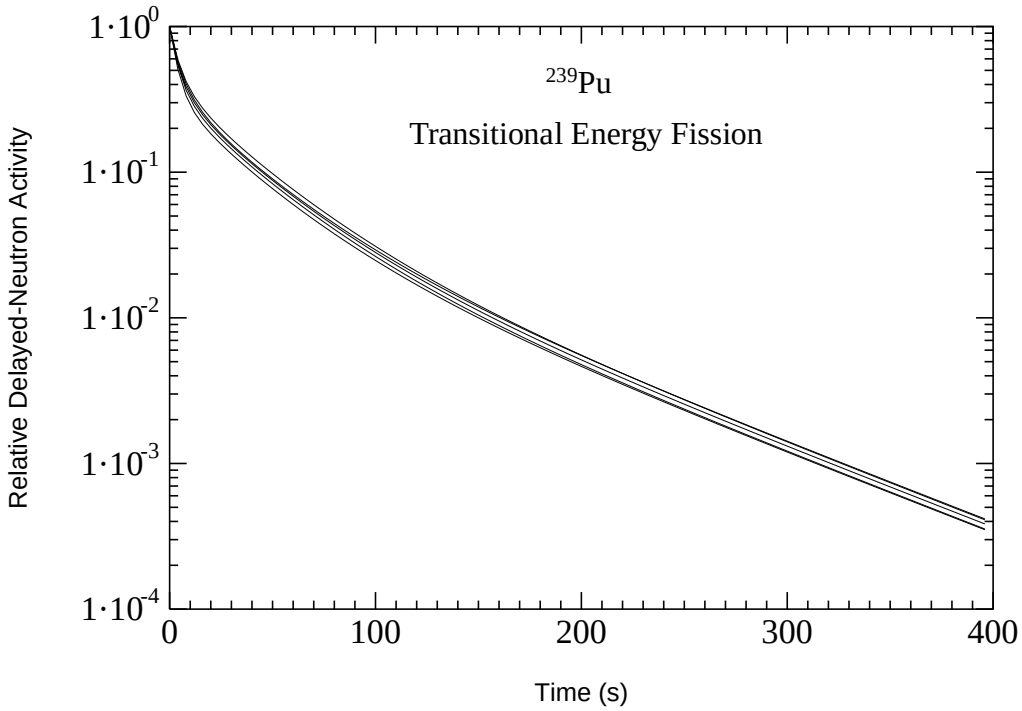

Fig. 29. Delayed Neutron Decay Curve for Transitional Energy Fission of ^{239}Pu.

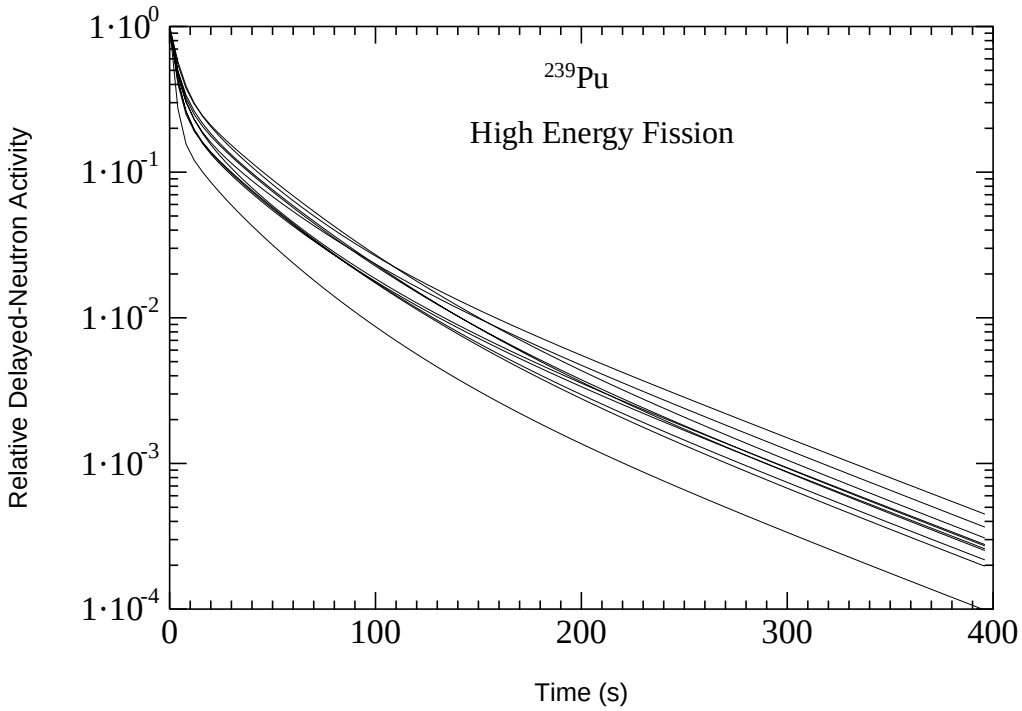

Fig. 30. Delayed Neutron Decay Curve for High Energy Fission of ^{239}Pu.

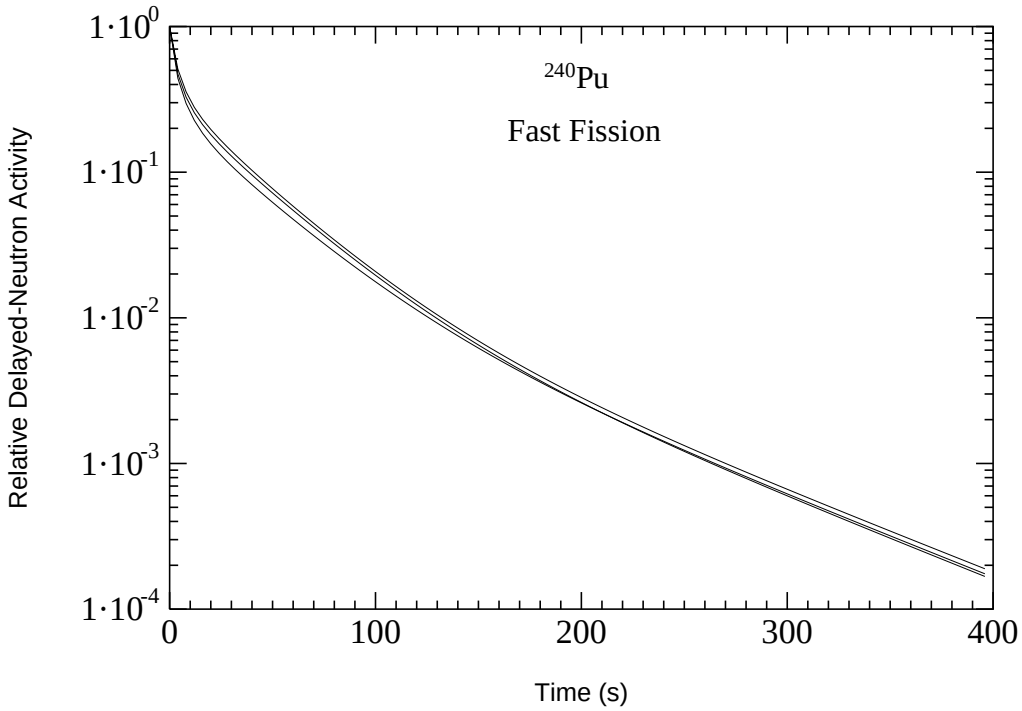

Fig. 31. Delayed Neutron Decay Curve for Fast Fission of ^{240}Pu.

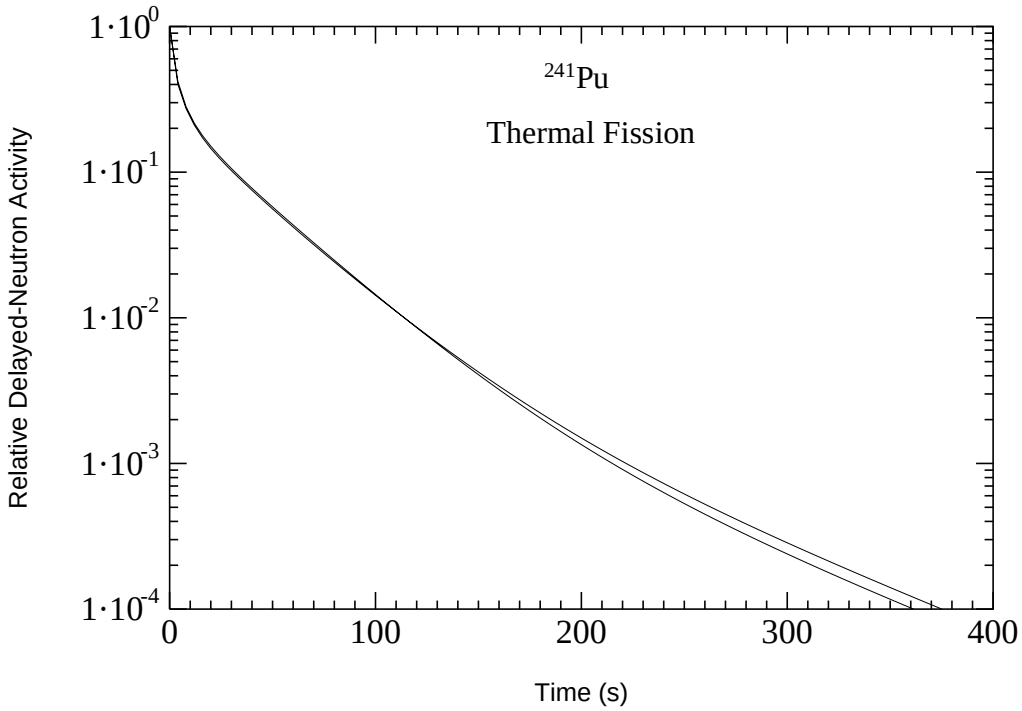

Fig. 32. Delayed Neutron Decay Curve for Thermal Fission of ^{241}Pu.

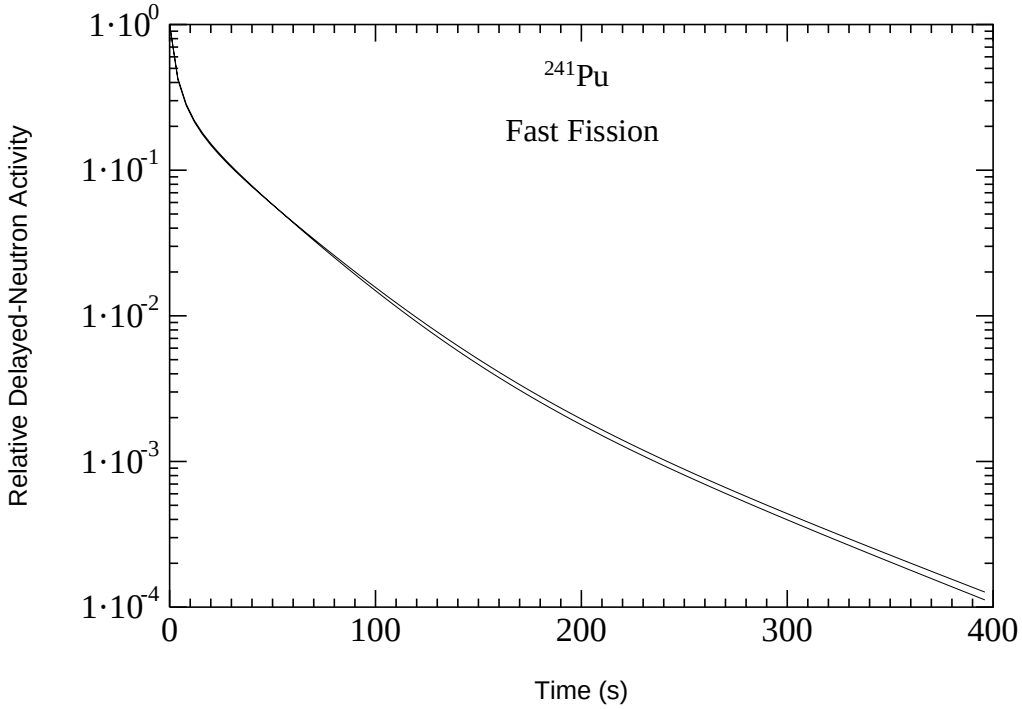

Fig. 33. Delayed Neutron Decay Curve for Fast Fission of ^{241}Pu.

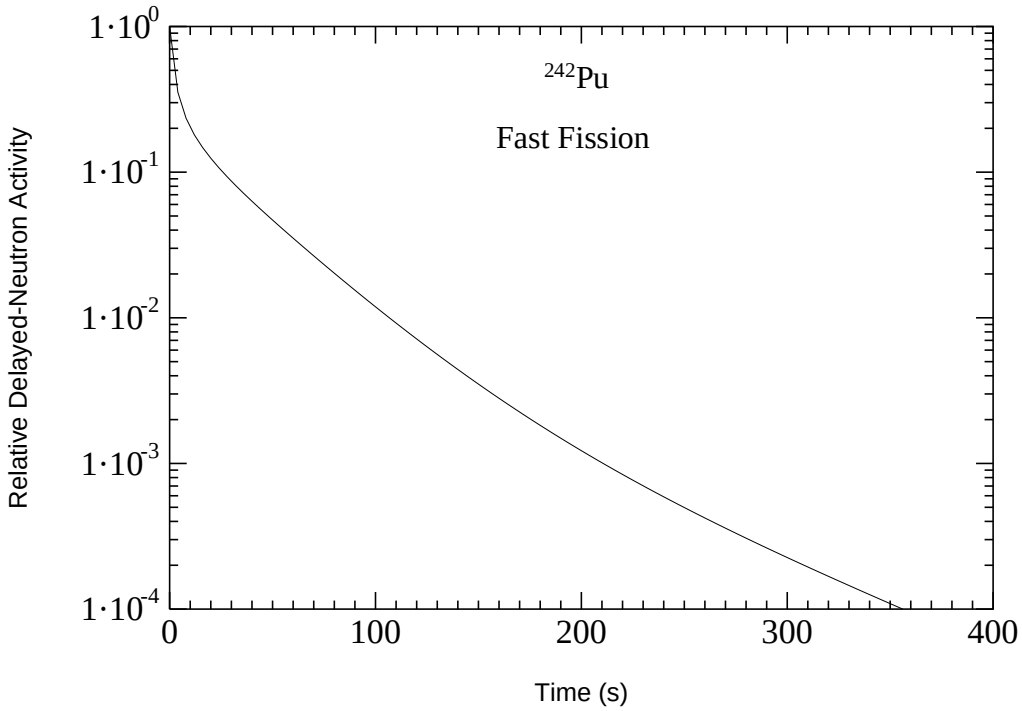

Fig. 34. Delayed Neutron Decay Curve for Fast Fission of ^{242}Pu.

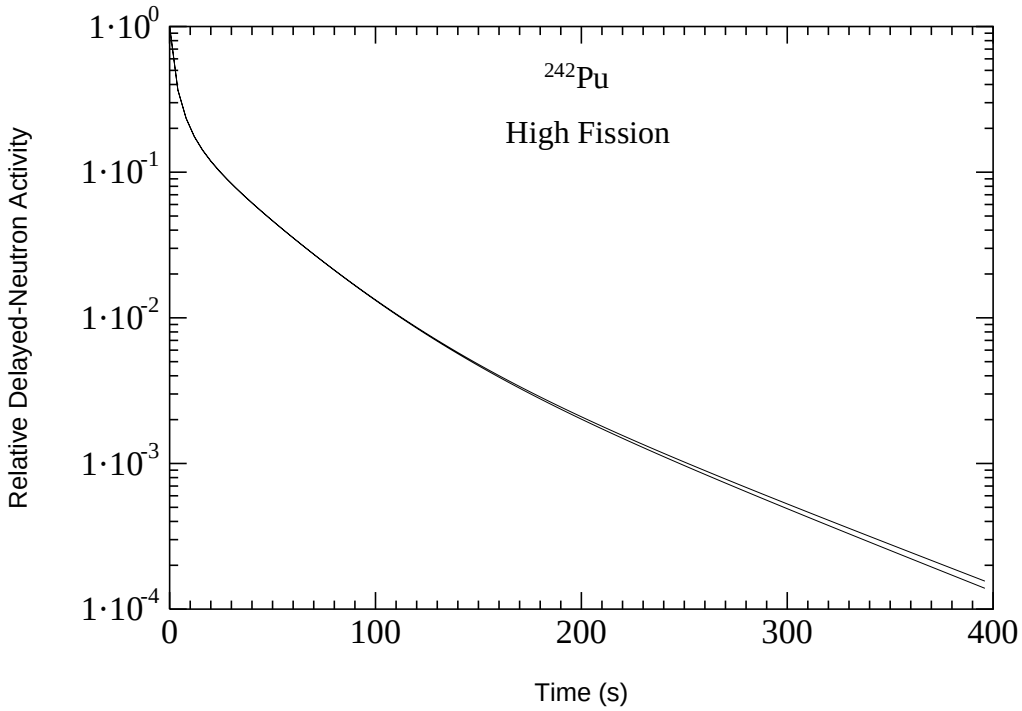

Fig. 35. Delayed Neutron Decay Curve for High Fission of ^{242}Pu.

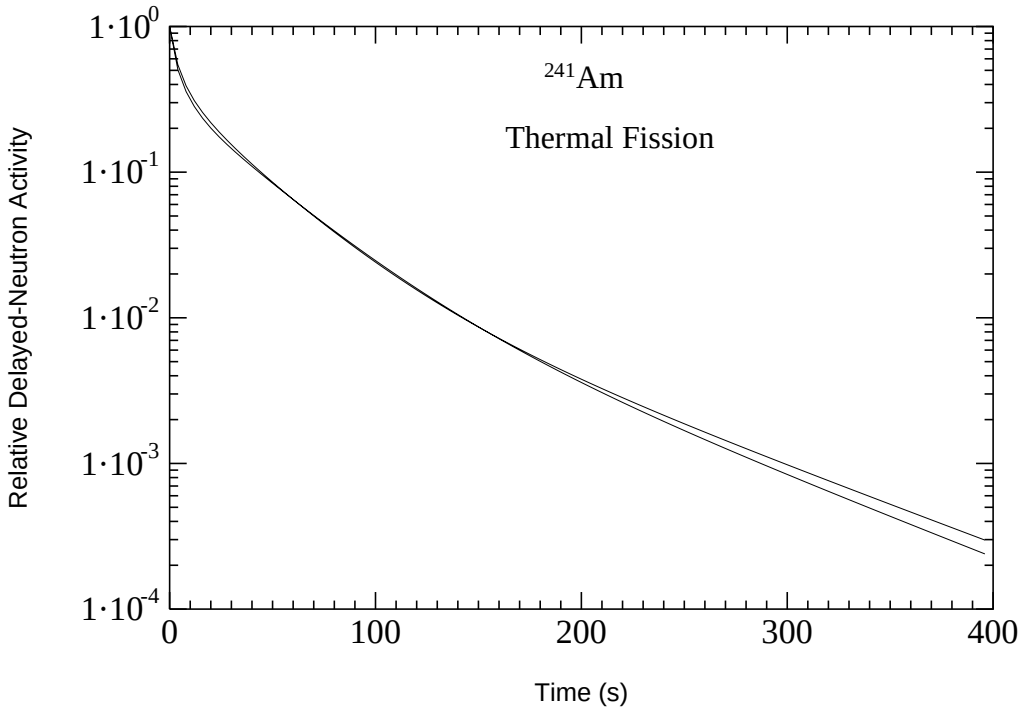

Fig. 36. Delayed Neutron Decay Curve for Thermal Fission of ^{241}Am.

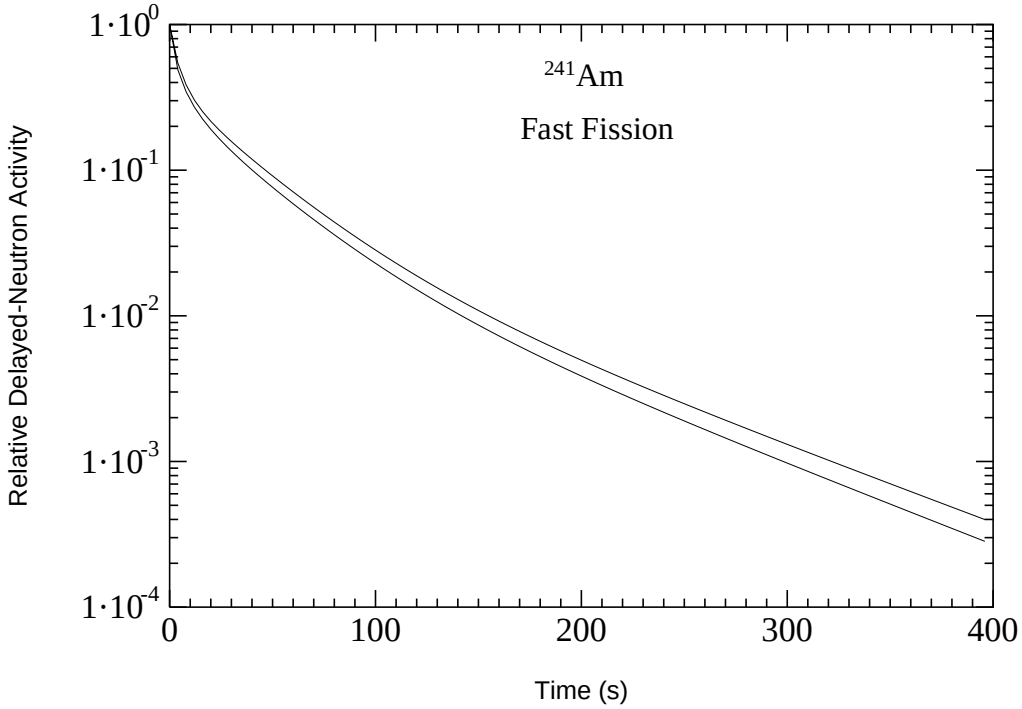

Fig. 37. Delayed Neutron Decay Curve for Fast Fission of ^{241}Am.

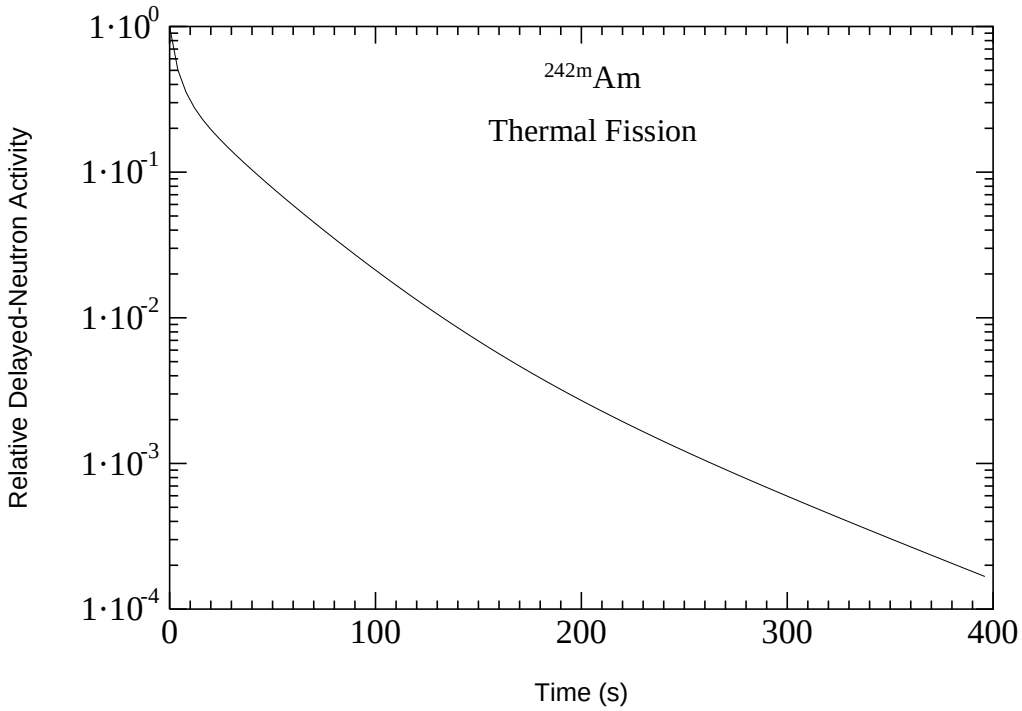

Fig. 38. Delayed Neutron Decay Curve for Thermal Fission of ^{242m}Am.

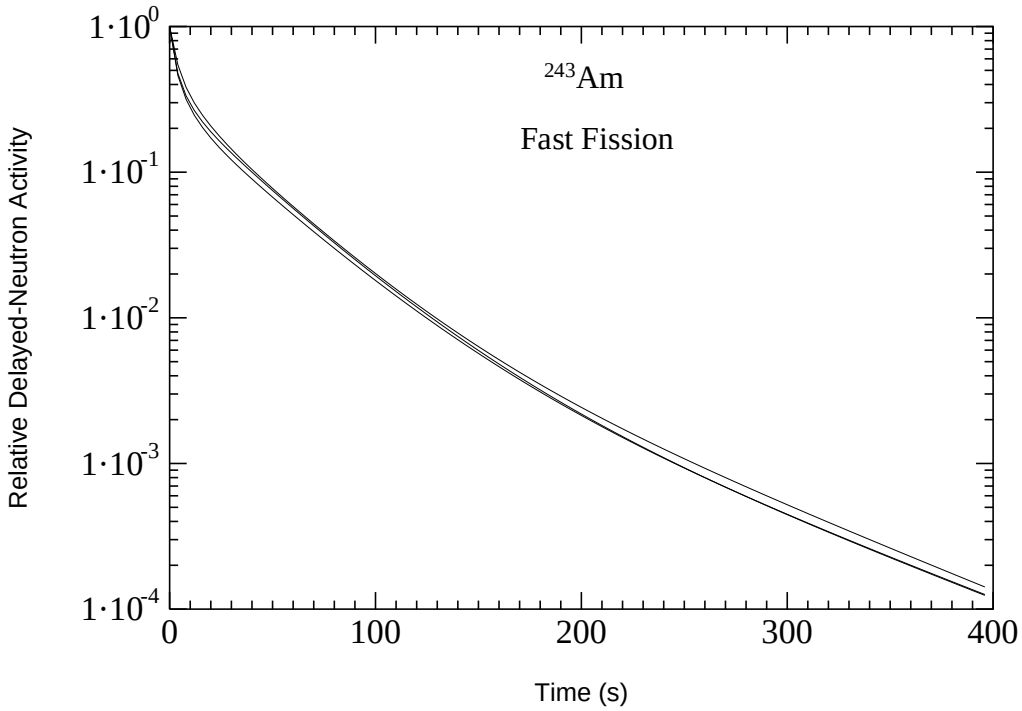

Fig. 39. Delayed Neutron Decay Curve for Fast Fission of ^{243}Am.

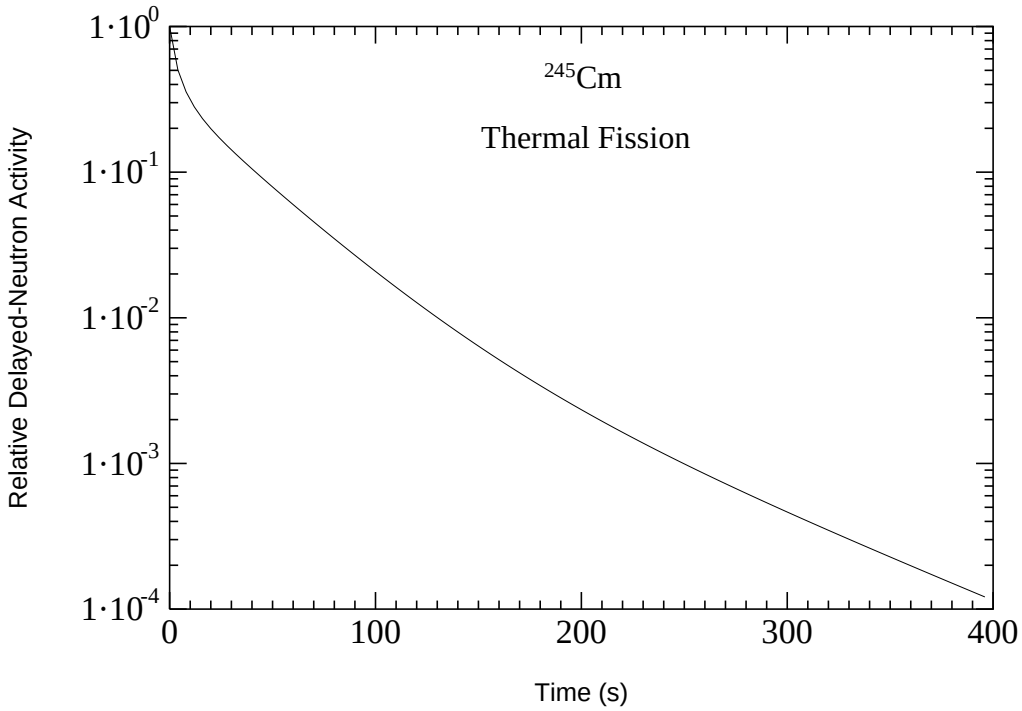

Fig. 40. Delayed Neutron Decay Curve for Thermal Fission of ^{245}Cm.

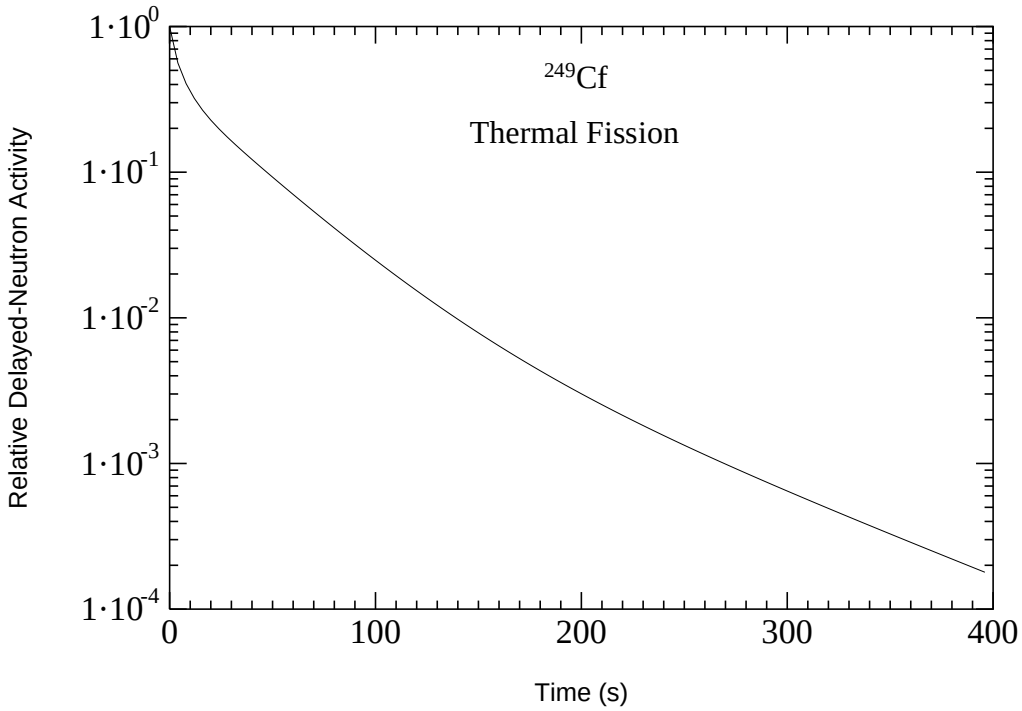

Fig. 41. Delayed Neutron Decay Curve for Thermal Fission of ^{249}Cf.

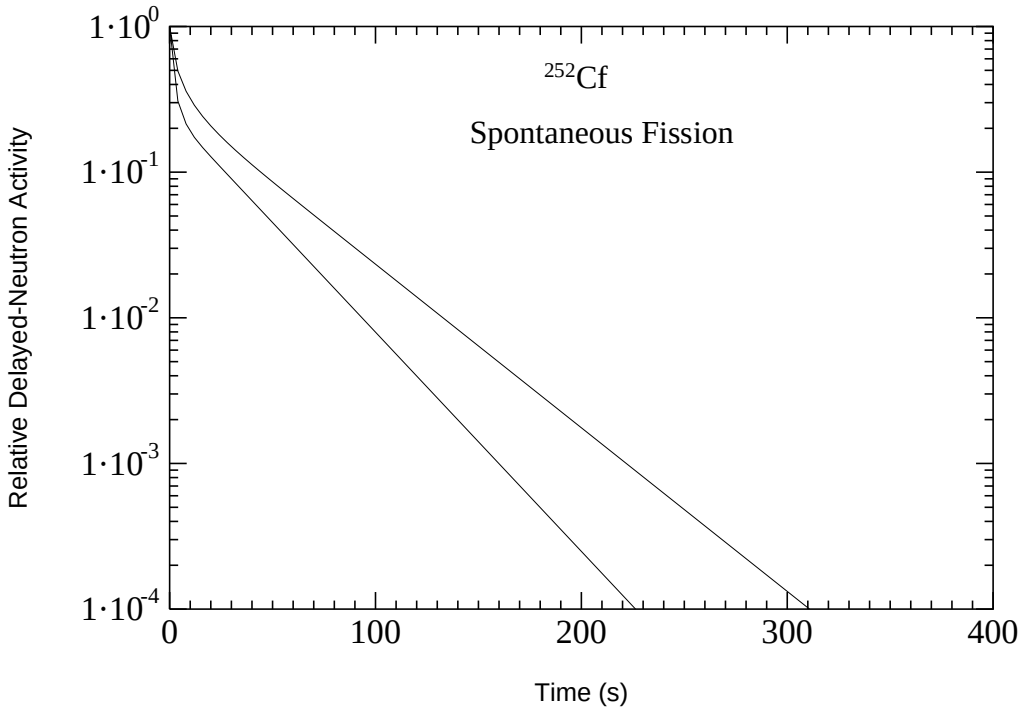

Fig. 42. Delayed Neutron Decay Curve for Spontaneous Fission of ^{249}Cf.

DISCUSSION

Delayed neutron parameters used in reactor dynamic calculations have traditionally been determined from a least-squares fit (LSF) of an aggregate decay curve of delayed neutrons emitted from a small sample of fissionable material irradiated by a strong neutron source. When performing the LSF, it has been customary to assume that the decay curve can be represented by the sum of exponentials (usually 5 or 6) in which both the abundances, a_i, and the decay constants, λ_i, of the exponentials are free parameters in the fit. As a consequence of allowing all parameters to be free in the LSF, the converged values of the abundances and decay constants usually differ from isotope-to-isotope as well as varying as a function of incident neutron energy. Furthermore, the decay constants obtained in this fashion will not necessarily converge to the decay constants of any of the 271 potential delayed-neutron emitters. As explained by Keepin (1965), the decay constants obtained during the LSF of the aggregate decay curve actually represent weighted averages based on the abundances and half-lives of the various precursors contributing to each group.

Although Keepin's rationale readily explains the variation in the decay constants for the short-lived and intermediate-lived groups, the observed variation in the decay constant of the longest-lived group (i.e., group 1 in the current 6-group model) is not as easily explained. Because ^{87}Br is the only known precursor of *significant* yield with a half-life on that order, then the half-life of group 1 is expected to correspond to the half-life of ^{87}Br regardless of the fissioning isotope or incident neutron energy. However, the measured half-life of group 1 exhibits rather large fluctuations (see Table IV) considering the precision of some of the individual measurements. We postulate that some of these fluctuations might be caused by unknown systematic errors associated with the data analysis technique.

One potential source of systematic error may be associated with the background correction at the tail end of the observed delayed neutron decay curve. In most of the delayed neutron experiments, the neutron background was reported to be on the order of 1 cps. Consequently, the uncertainty of the background count rate was quite large in all experiments except in those few cases in which the background was measured over long periods of time. When coupled with the large uncertainties associated with the small count rates encountered at the tail end of the decay curve (e.g., 2 or 3 cps above background), any small bias in the background correction could potentially cause the measured slope of the decay curve to differ from the true slope by a few percent.

Another potential source of systematic error is postulated to be associated with the maximum length of time the delayed neutron decay curves could be observed. The delayed neutron yields during these measurements were usually very small because of size limitations imposed on the samples to minimize multiplication effects. As a consequence, most experimenters were only able to follow their decay curves for approximately 300 s before reaching background. Unfortunately, this length of time is not quite long enough for the decay curves to reach their true asymptotic decay rate, which presumably corresponds to the decay constant of ^{87}Br. As an example, consider the decay curve for the thermal fissioning of ^{235}U generated using Keepin's six-group parameters. At 300 s the instantaneous decay rate is approximately 95% of the asymptotic decay rate, and does not reach 99.99% of the asymptotic decay rate until approximately 900 s. Hence, resolving the true asymptotic decay rate from an abbreviated decay curve (e.g., 0 to ~300 s) can be very challenging for most LSF algorithms, particularly when using data that exhibits large statistical fluctuations. To demonstrate this effect, a numerical test was performed using simulated data and the original least-squares-fitting code used by Keepin to fit his experimental data. When the simulated data spanned 1000 s, with most of the data occurring in the first 10 seconds of the decay curve, the LSF code was able to accurately resolve all of the decay constants and group abundances used to generate the simulated decay curve data. However, when

Table IV. Experimentally-Measured Half-Life of Group 1

Isotope	Half-life (s)	Isotope	Half-life (s)
Th-229_ther	55.72 ± 1.3	Np-237_fast	55.10 ± 0.18
Th-232_fast	56.03 ± 0.01	Np-237_fast	54.58 ± 0.86
Th-232_fast	55.41 ± 0.09	Np-237_fast	53.73 ± 2.5
U-232_fast	54.32 ± 0.17	Np-237_fast	55.32 ± 0.94
U-233_ther	55.60 ± 0.20	Np-237_inte	55.18 ± 0.55
U-233_ther	55.30 ± 0.90	Np-237_inte	54.65 ± 0.55
U-233_ther	55.00 ± 0.54	Np-237_inte	54.68 ± 0.55
U-233_ther	53.52 ± 37.	Np-237_inte	53.23 ± 0.53
U-233_ther	53.52 ± 37.	Pu-238_ther	54.92 ± 0.57
U-233_ther	55.94 ± 0.18	Pu-238_fast	49.51 ± 1.8
U-233_fast	55.11 ± 1.9	Pu-239_ther	55.00 ± 0.40
U-233_fast	54.15 ± 0.85	Pu-239_ther	54.28 ± 2.3
U-233_high	55.56 ± 0.40	Pu-239_ther	54.50 ± 43.
U-235_ther	57.00 ± 3.0	Pu-239_ther	54.50 ± 43.
U-235_ther	55.00 ± 0.40	Pu-239_ther	55.63 ± 0.05
U-235_ther	55.38 ± 0.69	Pu-239_fast	53.70 ± 3.6
U-235_ther	55.60 ± 0.20	Pu-239_fast	55.00 ± 0.87
U-235_ther	55.30 ± 0.90	Pu-239_fast	53.75 ± 0.95
U-235_ther	55.72 ± 1.3	Pu-240_fast	53.56 ± 1.2
U-235_ther	55.70 ± 1.9	Pu-240_fast	54.15 ± 1.3
U-235_ther	55.23 ± 0.13	Pu-241_ther	54.00 ± 1.0
U-235_ther	56.54 ± 0.55	Pu-241_ther	53.48 ± 0.41
U-235_ther	55.72 ± 1.3	Pu-241_fast	54.15 ± 1.3
U-235_ther	55.45 ± 4.0	Pu-242_fast	51.73 ± 1.0
U-235_fast	54.30 ± 0.90	Pu-242_high	53.70 ± 4.3
U-235_fast	53.32 ± 0.41	Pu-242_high	55.40 ± 4.5
U-235_fast	54.51 ± 0.94	Am-241_ther	54.54 ± 0.13
U-235_fast	54.58 ± 0.43	Am-241_fast	56.82 ± 0.93
U-235_fast	55.30 ± 0.82	Am-241_fast	56.82 ± 2.8
U-235_high	64.80 ± 11.	Am-42m_ther	54.45 ± 0.21
U-235_high	54.59 ± 0.50	Am-243_fast	52.91 ± 0.81
U-235_high	50.60 ± 1.9	Cm-245_ther	51.92 ± 0.35
U-238_fast	53.00 ± 1.7	Cf-249_ther	53.94 ± 0.08
U-238_high	56.31 ± 0.70		
Overall Average = 55.2 ± 0.03 s			

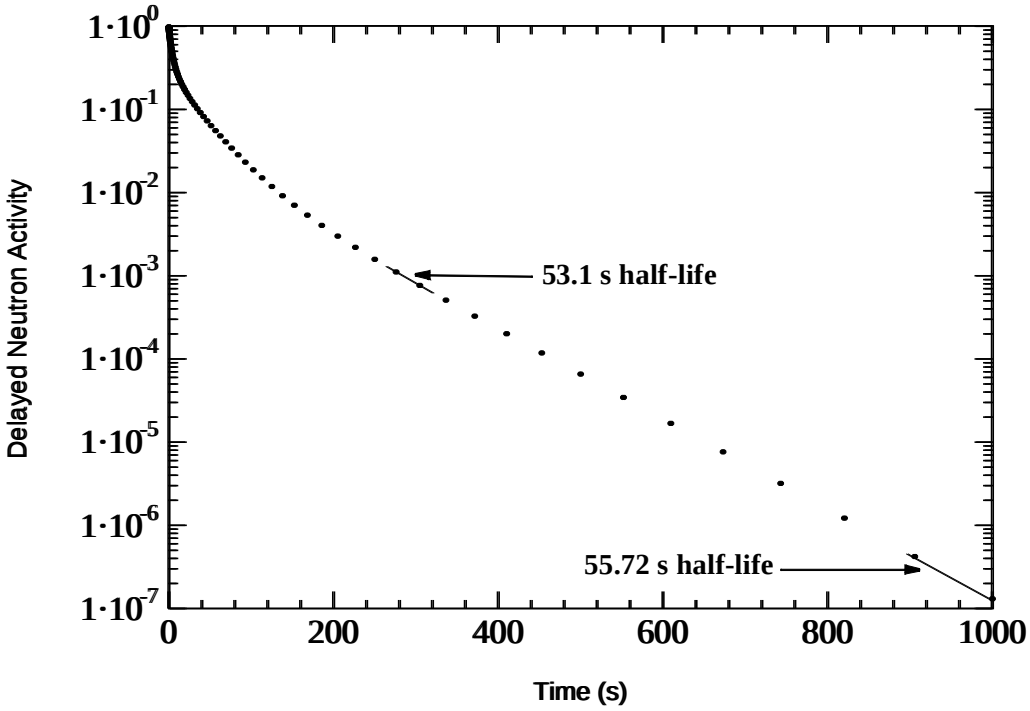

Fig. 43. Comparison of the instantaneous slope (expressed in terms of its corresponding half-life) of the delayed neutron decay curve for thermal fission of ^{235}U as measured by Keepin. At 300 s, the instantaneous slope is 53.1 s. At 900 s, the instantaneous slope is 55.72 s, which corresponds to the value used to generate the

the simulated data was truncated at 300 s and equally spaced in time to match real experimental data, the LSF code was unable to converge on all parameters. To force convergence, we found it necessary to fix the parameters for several of the short-lived groups at their correct value. However, this still did not guarantee that the remaining parameters would converge to their correct value. In particular, the decay constant for group 1 was consistently underestimated by approximately 2 to 5 %. This disparity is believed to be caused by *cross correlation* between the various parameters in the fit. That is to say, each parameter sought in the LSF is not truly independent of the other parameters in the model; the final value of each parameter is dependent, to some extent, on the final value of all of the other parameters. Hence, the truncated decay curve coupled with cross-correlation effects and potential biases in the background could possibly explain the large fluctuations in the group 1 decay constants shown in Table IV.

Notwithstanding the bias in the group-1 half-life, in most reactor systems the value of the half-life of group 1 has little effect on the reactivity scale for *positive* reactivities. However, the value of the group-1 half-life has a significant impact on the negative reactivity scale since the asymptotic root of the inhour equation is bounded by the decay constant of group 1. For example, in Table V, we compare the positive and negative reactivity scales for two different delayed neutron models: 1) Keepin's 6-group model for fast fission of ^{235}U, and 2) the equivalent 8-group model for Keepin's 6-group model derived during this study.[b] In Keepin's

b. To be discussed in another paper in this journal.

Table V. Comparison of Positive and Negative Reactivity Scales for Two Different Delayed Neutron Models.

Period (s)	54.5 s half-life[a]	55.6 s half-life[b]
.1000	.9619	.9620
.3000	.9050	.9051
1.000	.7817	.7817
3.000	.6080	.6080
10.00	.3908	.3907
30.00	.2212	.2212
100.0	.0963	.0963
300.0	.0380	.0380
1000.	.0123	.0123
-1000.	-.0132	-.0132
-500.0	-.0275	-.0275
-250.0	-.0605	-.0605
-125.0	-.1616	-.1621
-110.0	-.2097	-.2115
-100.0	-.2699	-.2749
-90.00	-.4133	-.4400
-85.00	-.6324	-.7500
-82.00	-1.0583	-1.7333

a. Keepin's 6-group model for fast fission of ^{235}U.
b. Equivalent 8-group model of Keepin's 6-group model for fast fission of ^{235}U. In the 8-group model, the group 1 half-life is specified at 55.6 s.

original 6-group model, the group-1 half-life is 54.5 s, whereas, in the equivalent 8-group model, the group-1 half-life is 55.6 s. As can be noted from this table, the positive reactivity scale is, for all intents and purposes, identical. However, as the asymptotic period approaches the asymptote associated with group 1, the negative reactivity scale is greatly effected by the value of the group-1 half-life.

For reasons that have never been thoroughly explained in the literature, it is commonly accepted that the reactivities inferred from negative period measurements are not as accurate as positive period measure-

ments. We now believe that this inaccuracy is largely attributable to the bias of the group-1 half-life from the half-life of ^{87}Br and, to a lesser extent, to the fact that the delayed neutrons from the next two longest-lived precursors, ^{137}I and ^{88}Br, are lumped into one group rather than being treated separately.

CONCLUSIONS

A literature survey of experimentally-measured delayed neutron parameters for 20 different fissionable isotopes has been performed. As a result, 245 sets of delayed neutron parameters have been identified. A comparison of the decay curves for each isotope as a function of the incident neutron energy has been performed and has shown that the results can be quite different.

From this study, it is concluded that more out-of-pile and in-pile experimental work is needed to clearly identify the delayed neutron parameters that produce the most accurate reactivity scale.

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